WCAP-17077-NP
Revision 0

July 2009

PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Robinson Nuclear Plant

Westinghouse

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2009

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LIST OF ACRONYMS

AMP	Aging Management Program
AMR	Aging Management Review
ARDM	age-related degradation mechanism
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
B&W	Babcock & Wilcox
BMI	bottom-mounted instrumentation
BWR	boiling water reactor
CASS	cast austenitic stainless steel
CE	Combustion Engineering
CES	Corporate Chief Engineering Section
CFR	Code of Federal Regulations
CLB	current licensing basis
CRGT	control rod guide tube
EFPY	effective full-power year
EPRI	Electric Power Research Institute
ET	electromagnetic testing (eddy current)
EVT	enhanced visual testing (a visual NDE method that includes EVT-1)
FMECA	failure mode, effects, and criticality analysis
GALL	Generic Aging Lessons Learned
I&E	Inspection and Evaluation
IASCC	irradiation-assisted stress corrosion cracking
IGSCC	intergranular stress corrosion cracking
INPO	Institute of Nuclear Power Operations
ISI	inservice inspection
ISR	irradiation-enhanced stress relaxation
LRA	License Renewal Application
MRP	Materials Reliability Program
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NES	Nuclear Engineering & Services
NOS	Nuclear Oversight Section
NRC	U.S. Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OBE	operating basis earthquake
OE	Operating Experience
OER	Operating Experience Report
PH	precipitation-hardenable (heat treatment)
PWR	pressurized water reactor
PWROG	Pressurized Water Reactor Owners Group (formerly WOG)
PWSCC	primary water stress corrosion cracking
QA	quality assurance
RCS	reactor coolant system
RI-FG	Reactor Internals Focus Group

LIST OF ACRONYMS (cont.)

RO	refueling outage
RNP	H. B. Robinson Steam Electric Plant, Unit Number 2, or Robinson Nuclear Plant
RV	reactor vessel
RVI	reactor vessel internals
SCC	stress corrosion cracking
SER	Safety Evaluation Report
SRP	Standard Review Plan
SS	stainless steel
UT	ultrasonic testing (a volumetric NDE method)
VT	visual testing (a visual NDE method that includes VT-1 and VT-3)
WOG	Westinghouse Owners Group
XL	Extra-long Westinghouse Fuel

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ACKNOWLEDGEMENTS

The authors would like to thank the members of the Progress Energy Aging Management Program Team lead by Charles Griffin and our associates at Westinghouse for their efforts in supporting development of this WCAP.

Progress Energy

1 PURPOSE

The purpose of this report is to document the H. B. Robinson Steam Electric Plant, Unit Number 2, hereafter Robinson Nuclear Plant (RNP), Reactor Vessel Internals (RVI) Aging Management Program (AMP) Plan. The purpose of the AMP is to manage the effects of aging on reactor vessel internals through the license renewal period. This document provides a description of the program as it relates to the management of aging effects identified in various regulatory and updated industry-generated documents in addition to the program documented in RNP calculation RNP-L/LR-0614 [8] in support of license renewal program evaluations. This AMP is supported by existing RNP documents and procedures and, as needed by industry experience or directive in the future, will be updated or supported by additional documents to provide clear and concise direction for the effective management of aging degradation in reactor internals components. These actions provide assurance that operations at RNP will continue to be conducted in accordance with the current licensing basis (CLB) for the reactor vessel internals by fulfilling License Renewal commitments [2], American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI Inservice Inspection (ISI) programs [17], and industry requirements [11]. This AMP fully captures the intent of the additional industry guidance for reactor internals augmented inspections, based on the programs sponsored by U.S. utilities through the Electric Power Research Institute (EPRI) managed Materials Reliability Program (MRP) and the Pressurized Water Reactor Owners Group (PWROG).

The main objectives for the RNP RVI AMP are to:

- Demonstrate that the effects of aging on the RVI will be adequately managed for the period of extended operation in accordance with 10 CFR 54 [1].
- Summarize the role of existing RNP AMPs in the RVI AMP.
- Define and implement the industry-defined (EPRI/MRP and PWROG) pressurized water reactor (PWR) RVI requirements and guidance for managing aging of reactor internals.
- Provide an inspection plan summary for the RNP reactor internals.

RNP License Renewal Commitment 33 [2], "Pressurized Water Reactor Vessel Internals Program," commits RNP to:

1. Participate in industry programs to investigate aging effects and determine the appropriate AMP activities to address baffle and former assembly issues, and to address changes in dimensions due to void swelling.

2. Evaluate the results of completed research projects from the Westinghouse Owners Group (formerly WOG, now PWROG) and the EPRI MRP, and factor them into the PWR Vessel Internals Program as appropriate.

3. Implement an augmented inspection during the license renewal term.

Augmented inspections, based on required program enhancements resulting from industry programs, will become part of the RNP ASME B&PV Code, Section XI program [17]. Corrective actions for augmented inspections will be developed using repair and replacement procedures equivalent to those requirements in ASME B&PV Code, Section XI, or as determined independently by Progress Energy, or in cooperation with the industry, to be equivalent or more rigorous than currently defined procedures.

This AMP for the RNP reactor internals demonstrates that the program adequately manages the effects of aging for reactor internals components and establishes the basis for providing reasonable assurance that the internals components will continue to perform their intended function through the RNP license renewal period of extended operation. It also supports the RNP license renewal commitment to submit to the U.S. Nuclear Regulatory Commission (NRC) an inspection plan for the PWR Vessel Internals Program, as it will be implemented from RNP's participation in industry initiatives, 24 months prior to the augmented inspection.

The development and implementation of this program meets the license renewal commitment.

2 BACKGROUND

The management of aging degradation effects in reactor internals is required for nuclear plants considering or entering license renewal, as specified in the NRC Standard Review Plan [3]. The U.S. nuclear power industry has been actively engaged in recent years in a significant effort to support the industry goal of responding to these requirements. Various programs have been underway within the industry over the past decade to develop guidelines for managing the effects of aging within PWR reactor internals. In 1997, the WOG issued WCAP-14577 [27], "License Renewal Evaluation: Aging Management for Reactor Internals," which was reissued as Revision 1-A in 2001 after receiving NRC Staff review and approval. Later, an effort was engaged by the EPRI MRP to address the PWR internals aging management issue for the three currently operating U.S. reactor designs – Westinghouse, Combustion Engineering (CE), and Babcock & Wilcox (B&W).

The MRP first established a framework and strategy for the aging management of PWR internals components using proven and familiar methods for inspection, monitoring, surveillance, and communication. Based upon that framework and strategy, and on the accumulated industry research data, the following elements of an Aging Management Program were further developed [20, 27, 33]:

- Screening criteria were developed, considering chemical composition, neutron fluence exposure, temperature history, and representative stress levels, for determining the relative susceptibility of PWR internals components to each of eight postulated aging mechanisms (further discussed in Section 4 of this Program).

- PWR internals components were categorized, based on the screening criteria, into categories that ranged from:

  - Components for which the effects from the postulated aging mechanisms are insignificant,
  - Components that are moderately susceptible to the aging effects, and
  - Components that are significantly susceptible to the aging effects.

- Functionality assessments were performed, based on representative plant designs of PWR internals components and assemblies of components using irradiated and aged material properties, to determine the effects of the degradation mechanisms on component functionality.

Aging management strategies were developed combining the results of functionality assessment with several contributing factors to determine the appropriate aging management methodology, baseline examination timing, and the need and timing of subsequent inspections. Items considered included component accessibility, operating experience, existing evaluations, and prior examination results.

The industry effort, as coordinated by the EPRI MRP, has finalized initial Inspection and Evaluation Guidelines (I&E Guidelines) for reactor internals and submitted the document to the NRC with a request for a formal Safety Evaluation Report (SER). A supporting document addressing inspection requirements is also being completed. The industry guidance is contained within two separate EPRI MRP documents:

- MRP-227, "PWR Internals Inspection and Evaluation Guidelines," [11] (hereafter referred to as "the I&E Guidelines" or simply "MRP-227") provides the industry background, listing of reactor internals components requiring inspection, type of NDE required for each component, timing for initial inspections, and criteria for evaluating inspection results. MRP-227 provides a standardized approach to PWR internals aging management for each unique reactor design (Westinghouse, B&W, and CE). The document was submitted to the NRC for a formal evaluation and review.

- MRP-228 [28], "Inspection Standard for Reactor Internals Components," provides guidance on the qualification/demonstration of the NDE techniques and other criteria pertaining to the actual performance of the inspections.

The PWROG has also recently begun efforts to develop "generic acceptance criteria" for the MRP-227 inspections, where feasible, for some of the reactor internals components. Final reports are to be developed and be available for industry use in support of planned license renewal inspection commitments. In some cases, individual plants will develop plant-specific acceptance criteria for some internals components where a generic approach is not practical.

The RNP reactor internals are integral with the reactor coolant system (RCS) of a Westinghouse three-loop nuclear steam supply system (NSSS), a typical illustration of which is provided in Figure A-1.

As described in NUREG-1785 [2], the RNP internals are designed to support, align, and guide the core components and to support and guide in-core instrumentation. The RVI consist of two basic assemblies – an upper internals assembly that is removed during each refueling operation to obtain access to the reactor core, and a lower internals assembly that can be removed, if desired, following a complete core unload.

The lower internals assembly is supported in the vessel by resting on a ledge in the vessel head-mating surface and is closely guided at the bottom by radial support/clevis assemblies. The upper internals assembly is clamped at this same ledge by the reactor vessel head. The bottom of the upper internals assembly is closely guided by the core barrel alignment pins of the lower internals assembly.

The lower internals comprise the core barrel, thermal shield, core baffle assembly, lower core plate, intermediate diffuser plate, bottom support plate, and supporting structures. The upper internals package (upper core support structure) is a rigid member composed of the top support plate and deep beam sections, support columns, control rod guide tube assemblies, and the upper core plate. Upon upper internals assembly installation, the last three parts are physically located inside the core barrel.

The in-core instrumentation includes in-core flux guide thimbles to permit the insertion of movable detectors for measurement of the neutron flux distribution within the reactor core. Movable miniature neutron flux detectors are available to scan the active length of selected fuel assemblies to provide remote reading of the relative three-dimensional flux distribution. The thimbles are inserted into the reactor core through guide tubes, or conduits, extending from the bottom of the reactor vessel (RV) through the concrete shield area and then up to a thimble seal table. Since the movable detector thimbles are closed at the leading (reactor) end, they are dry inside. The thimbles thus serve as a pressure barrier between the reactor coolant pressure and the atmosphere. Mechanical seals between the retractable thimbles and the conduits are provided at the seal table.

RNP was granted a license for extended operation by the NRC through the issuance of a SER in NUREG-1785 [2]. In the SER, the NRC concluded that the RNP License Renewal Application (LRA) adequately identified the RVI system structures and components that are subject to an AMR, as required by 10 CFR 54.21(a)(1) [1]. A listing of the RNP reactor vessel internals components and subcomponents already reviewed by the NRC in the SER that are subject to AMP requirements is included in Tables B-1 and B-2.

In accordance with 10 CFR Part 54 [1], frequently referred to as the License Renewal Rule, RNP has developed a procedure to direct the performance of aging management reviews of mechanical structures and components [9]. The U.S. industry, as noted through the efforts of the MRP and PWROG, has further investigated the components and subcomponents that require aging management to support continued reliable function. As designated by the protocols of NEI 03-08 [12], "Guidelines for the Management of Materials Issues," each plant will be required to use MRP-227 and MRP-228 to develop and implement an AMP for reactor internals no later than three years after the initial industry issuance of MRP-227. MRP-227 was issued in December 2008, and plant AMPs must therefore be completed by December 2011, or sooner, if required by plant-specific License Renewal Commitments.

The information contained in this AMP fully complies with the requirements and guidance of the referenced documents. The AMP will manage aging effects of the RVI so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

3 PROGRAM OWNER

The PWR Vessel Internals Program is part of the "Progress Energy Reactor Coolant System Material Integrity Management Program" [7]. The successful implementation and comprehensive long-term management of the RNP RVI AMP will require the integration of Progress Energy organizations, corporately and at RNP, and interaction with multiple industry organizations including, but not limited to, the ASME, MRP, NRC, and PWROG. The responsibilities of the individual Progress Energy corporate and RNP groups are provided in the following paragraphs. Progress Energy will maintain cognizance of industry activities related to PWR internals inspection and aging management, and will address/implement industry guidance stemming from those activities, as appropriate under NEI 03-08 practices.

The overall responsibility for administration of the RVI AMP is RNP Senior Management.

Additional responsibilities and the appropriate responsible personnel are discussed in the following subsections.

3.1 CORPORATE NUCLEAR ENGINEERING SERVICES CHIEF ENGINEERING SECTION

The Nuclear Engineering Services (NES) Chief Engineering Section (CES) is responsible for providing governance and oversight of the implementation of the PWR Vessel Internals Program, including:

- Ensuring that appropriate programs are established and maintained to support inspection and mitigation activities for the reactor internals.
- Providing oversight of plant implementation of inspection and mitigation activities.
- Maintaining cognizance of industry activities related to PWR internals inspection and aging management.

3.2 RNP ENGINEERING

RNP Engineering is responsible for the overall development and implementation of the PWR Vessel Internals Aging Management Program, including:

- Planning, control, and implementation of the RVI AMP mitigation, inspection, and repair activities, as approved by site senior management.
- Review and approval of vendor programs involved in various reactor internals activities.
- Ensuring that required inspections and supporting activities are implemented in the times specified.

3.3 RNP ENVIRONMENTAL AND CHEMISTRY

RNP Environmental and Chemistry is responsible for:

- Maintaining primary water chemistry in accordance with approved RNP procedures and specifications.

- Participation in industry activities addressing water chemistry issues as they relate to minimizing the potential initiation and growth of primary water stress corrosion cracking (PWSCC) in nickel-based alloys and intergranular stress corrosion cracking (IGSCC) in austenitic stainless steel components.

4 DESCRIPTION OF THE H. B. ROBINSON REACTOR INTERNALS AGING MANAGEMENT PROGRAMS AND INDUSTRY PROGRAMS

The U.S. nuclear industry, through the combined efforts of utilities, vendors, and independent consultants, has defined a generic guideline to assist utilities in developing reactor internals plant-specific aging management programs based on inspection and evaluation. As noted in Section 3 of this AMP, the PWR Vessel Internals Program is a part of the "Progress Energy Reactor Coolant System Material Integrity Management Program" [7]. The intent of this program is to ensure the long-term integrity and safe operation of the reactor internals components. RNP has developed this AMP in conformance with the 10 Generic Aging Lessons Learned (GALL) [4] attributes and MRP-227.

This reactor internals AMP utilizes a combination of prevention, mitigation, and condition monitoring. Where applicable, credit is taken for existing programs such as water chemistry [15, 20], inspections prescribed by the ASME Section XI Inservice Inspection Program [17], thimble tube inspections [18], and past and future mitigation projects such as split pin replacement [26], combined with augmented inspections or evaluations as recommended by MRP-227.

Aging degradation mechanisms that impact internals have been identified and documented in RNP Aging Management Reviews [5, 6] prepared using the corporate procedural guidance document [9] in support of the License Renewal effort. The overall outcome of the reviews and the additional work performed by the industry, as summarized in MRP-227, is to provide appropriate augmented inspections for reactor internals components to provide early detection of the degradation mechanisms of concern. Therefore, this AMP is consistent with the existing RNP AMR methodology and the additional industry work summarized in MRP-227. All sources are consistent and address concerns about component degradation resulting from the following eight material aging degradation mechanisms identified as affecting reactor internals:

- Stress Corrosion Cracking (SCC)

  Stress corrosion cracking (SCC) refers to local, nonductile cracking of a material due to a combination of tensile stress, environment, and metallurgical properties. The actual mechanism that causes SCC involves a complex interaction of environmental and metallurgical factors. The aging effect is cracking.

- Irradiation-Assisted Stress Corrosion Cracking

  Irradiation-assisted stress corrosion cracking (IASCC) is a unique form of SCC that occurs only in highly irradiated components. The aging effect is cracking.

- Wear

  Wear is caused by the relative motion between adjacent surfaces, with the extent determined by the relative properties of the adjacent materials and their surface condition. The aging effect is loss of material.

- Fatigue

  Fatigue is defined as the structural deterioration that can occur as the result of repeated stress/strain cycles caused by fluctuating loads and temperatures. After repeated cyclic loading of sufficient magnitude, microstructural damage can accumulate, leading to macroscopic crack initiation at the most highly affected locations. Subsequent mechanical or thermal cyclic loading can lead to growth of the initiated crack. Corrosion fatigue is included in the degradation description.

  Low-cycle fatigue is defined as cyclic loads that cause significant plastic strain in the highly stressed regions, where the number of applied cycles is increased to the point where the crack eventually initiates. When the cyclic loads are such that significant plastic deformation does not occur in the highly stressed regions, but the loads are of such increased frequency that a fatigue crack eventually initiates, the damage accumulated is said to have been caused by high-cycle fatigue. The aging effects of low-cycle fatigue and high-cycle fatigue are additive.

  Fatigue crack initiation and growth resistance are governed by a number of material, structural, and environmental factors such as stress range, loading frequency, surface condition, and presence of deleterious chemical species. Cracks typically initiate at local geometric stress concentrations such as notches, surface defects, and structural discontinuities. The aging effect is cracking.

- Thermal Aging Embrittlement

  Thermal aging embrittlement is the exposure of delta ferrite within cast austenitic stainless steel (CASS) and precipitation-hardenable (PH) stainless steel to high inservice temperatures, which can result in an increase in tensile strength, a decrease in ductility, and a loss of fracture toughness. Some degree of thermal aging embrittlement can also occur at normal operating temperatures for CASS and PH stainless steel internals. CASS components have a duplex microstructure and are particularly susceptible to this mechanism. While the initial aging effect is loss of ductility and toughness, unstable crack extension is the eventual aging effect if a crack is present and the local applied stress intensity exceeds the reduced fracture toughness.

- Irradiation Embrittlement

  Irradiation embrittlement is also referred to as neutron embrittlement. When exposed to high-energy neutrons, the mechanical properties of stainless steel and nickel-based alloys can be changed. Such changes in mechanical properties include increasing yield strength, increasing ultimate strength, decreasing ductility, and a loss of fracture toughness. The irradiation embrittlement aging mechanism is a function of both temperature and neutron fluence. While the initial aging effect is loss of ductility and toughness, unstable crack extension is the eventual aging effect if a crack is present and the local applied stress intensity exceeds the reduced fracture toughness.

- Void Swelling and Irradiation Growth

  Void swelling is defined as a gradual increase in the volume of a component caused by formation of microscopic cavities in the material. These cavities result from the nucleation and growth of clusters of irradiation-produced vacancies. Helium produced by nuclear transmutations can have a significant impact on the nucleation and growth of cavities in the material. Void swelling may produce dimensional changes that exceed the tolerances on a component. Strain gradients produced by differential swelling in the system may produce significant stresses. Severe swelling (>5 percent by volume) has been correlated with extremely low fracture toughness values. Also included in this mechanism is irradiation growth of anisotropic materials, which is known to cause significant dimensional changes within in-core instrumentation tubes that are fabricated from zirconium alloys. While the initial aging effect is dimensional change and distortion, severe void swelling may result in cracking under stress.

- Thermal and Irradiation-Enhanced Stress Relaxation or Irradiation-Enhanced Creep

  The loss of preload aging effect can be caused by the aging mechanisms of stress relaxation or creep. Thermal stress relaxation (or primary creep) is defined as the unloading of preloaded components due to long-term exposure to elevated temperatures, as seen in PWR internals. Stress relaxation occurs under conditions of constant strain where part of the elastic strain is replaced with plastic strain. Available data show that thermal stress relaxation appears to reach saturation in a short time (< 100 hours) at PWR internals temperatures.

  Creep (or more precisely, secondary creep) is a slow, time- and temperature-dependent, plastic deformation of materials that can occur at stress levels below the yield strength (elastic limit). Creep occurs at elevated temperatures where continuous deformation takes place under constant strain. Secondary creep in austenitic stainless steels is associated with temperatures higher than those relevant to PWR internals even after taking into account gamma heating. However, irradiation-enhanced creep (or more simply, irradiation creep) or irradiation-enhanced stress relaxation (ISR) is an athermal process that depends on the neutron fluence and stress, and it can also be affected by void swelling should it occur. The aging effect is a loss of mechanical closure integrity (or preload) that can lead to unanticipated loading that, in turn, may eventually cause subsequent degradation by fatigue or wear and result in cracking.

The RNP RVI AMP is focused on meeting the requirements of the 10 elements of an aging management program as described in NUREG-1801, GALL Report Section XI.M16 for PWR Vessel Internals. In the RNP RVI AMP, this is demonstrated through application of existing RNP AMR methodology that credits inspections prescribed by the ASME Section XI Inservice Inspection Program, existing RNP programs, and additional augmented inspections based on MRP-227 recommendations. A description of the applicable existing RNP programs and compliance with the elements of the GALL is contained in the following subsections.

4.1 EXISTING RNP PROGRAMS

RNP's overall strategy for managing aging in reactor internals components is supported by the following existing programs:

- Primary Water Chemistry Program
- ASME Section XI Inservice Inspection Program
- Flux Thimble Eddy Current Program

These are established programs that support the aging management of RCS components in addition to the RVI components. Although affiliated with and supporting the RVI AMP, they will be managed under the existing programs.

Brief descriptions of the programs are included in the following subsections.

4.1.1 Primary Water Chemistry Program

The RNP Primary Water Chemistry Program [16] is used to mitigate aging effects on component surfaces that are exposed to water as process fluid. Chemistry programs are used to control water chemistry for impurities that accelerate corrosion and contaminants that may cause cracking due to SCC. This program relies on monitoring and control of water chemistry to keep peak levels of various contaminants below the system-specific limits. The RNP Primary Water Chemistry Program is based on the current revision of EPRI PWR Primary Water Chemistry Guidelines [20]. Later revisions of the guidelines will be used when issued. The limits imposed by the RNP program meet the intent of the industry standard for addressing primary water chemistry [20].

The evaluation of this program against the 10 attributes in the GALL for Program XI.M2 in support of the RNP LRA remains applicable.

4.1.2 ASME Section XI Inservice Inspection Program

The RNP ASME Code Section XI, ISI Program [17] is implemented to monitor for aging effects such as cracking, loss of preload due to stress relaxation or irradiation creep, loss of material, and reduction of fracture toughness due to thermal embrittlement. For RNP, inspections conducted under the reactor internals AMPs will be controlled as a combination of ASME Section XI ISI exams on core support structures and augmented exams performed under that ISI Program for the remaining reactor internals components addressed within MRP-227. The RNP Section XI, 10-year ISI examinations supporting the license renewal period are currently scheduled for fall 2011, RO-27.

The evaluation of this program against the 10 attributes in the GALL for Program XI.M1 in support of the RNP LRA remains applicable.

4.1.3 Flux Thimble Eddy Current Program

Flux thimble tubes are long, slender, stainless steel tubes that are seal welded at one end with flux thimble tube plugs, which pass through the vessel penetration, through the lower internals assembly, and finally extend to the top of the fuel assembly. The bottom-mounted instrumentation (BMI) column assemblies provide a path for the flux thimbles into the core from the bottom of the vessel and protect the flux thimbles during operation of the reactor. The flux thimble provides a path for the neutron flux detector into the core and is subject to reactor coolant pressure on the outside and containment pressure on the inside.

The RNP thimble tube inspection program is an existing plant-specific program that satisfies NRC Bulletin 88-09 requirements that a tube wear inspection procedure be established [18] and maintained for Westinghouse-supplied reactors that use bottom-mounted flux thimble tube instrumentation. The program follows Branch Technical Position RLSB-1, Aging Management Review – Generic, which is included in Appendix A of NUREG-1800. The program includes eddy current testing requirements for thimble tubes and criteria for determining sample size, inspection frequency, flaw evaluation, and corrective action in accordance with NRC Bulletin 88-09 [21].

The evaluation of this program against the 10 attributes in the GALL for Program XI.M16 in support of the RNP LRA remains applicable.

4.2 SUPPORTING RNP PROGRAMS AND AGING MANAGEMENT SUPPORTIVE PLANT ENHANCEMENTS

4.2.1 Reactor Internals Aging Management Review Process

A comprehensive review of aging management of reactor internals was performed according to the requirements of the License Renewal Rule [1] as directed by corporate procedure EGR-NGGC-0504, "Mechanical System Aging Management Review for License Renewal" [9]. The corporate procedure directs the use of calculations as the vehicle to implement the license renewal aging management review and to document the results. Calculation RNP-L/LR-0354A [5] and RNP-L/LR-0354B [6] document the results of the aging management review performed in support of RNP license renewal for reactor internals. The RNP LRA was approved by the NRC in NUREG-1785 [2]. RVI components specifically noted as requiring aging management, as identified in the NUREG, are summarized in Appendix B Tables B-1 and B-2 of this AMP.

The calculations supported the LRA as follows:

1. Identified applicable aging effects requiring management

2. Associated aging management programs to manage those aging effects

3. Identified enhancements or modifications to existing programs, new aging management programs, or any other actions required to support the conclusions reached in the calculation

Aging management reviews were performed for each RNP system that contained long-lived, passive components requiring aging management review, in accordance with the screening process of RNP EGR-NGGC-0503, "Mechanical Component Screening for License Renewal" [22]. This review is not repeated here, but the results are fully incorporated into the RNP RVI AMP.

4.2.2 Flux Thimble Tubes

A comprehensive evaluation was performed to evaluate the expected wear for the BMI combination thimble tubes at RNP, and the results are documented in WCAP-12202 [24]. As a result of the evaluation and subsequent operation, a plant modification was performed. The modification consisted of a cut-and-cap operation to secure thimbles N-5 and N-12 into their respective conduits at the seal table as a result of failure to achieve full insertion of the components. Following the initial modification, the remaining portions of thimbles N-5 and N-12 were removed through the core in a subsequent refueling outage with the final configuration being two completely removed thimbles.

4.2.3 Reactor Head and Control Rod Drive Mechanism Replacement

The problems associated with degradation of Alloy 600 materials and PWSCC are well known and documented in the industry. As a result of the identification of the existence of the concern at RNP, a program was undertaken to replace the undesirable material with one more resistant to the effects of PWSCC for the reactor head. A side benefit noted to the selected component replacement option was the introduction of design features to facilitate refueling outage activities. Detailed descriptions of all facets of the replacement are retained in the plant records.

4.2.4 Control Rod Guide Tube Split Pin Replacement Project

The control rod guide tube support pins are used to align the bottom of the control rod guide tube assembly into the top of the core plate. In general, SCC prevention is aided by adherence to strict primary water chemistry limits that effectively prevent SCC and greatly reduce the probability of IASCC. The limits imposed by the Primary Water Chemistry Program at RNP are consistent with the latest EPRI guidelines as described in Section 4.1.

The original RNP support pins were fabricated from INCONEL® alloy X-750 that was hot rolled, solution treated or annealed, and age hardened at various temperatures and times depending on heat, manufacturer, and fabrication date. Support pins made of this material with the associated heat treatments were shown to be susceptible to IGSCC and likely to fail during the lifetime of a nuclear power plant. Westinghouse developed an improved support pin design and fabrication technique that significantly reduced the susceptibility to IGSCC while maintaining the fatigue and wear requirements necessary to support continued uninterrupted service.

In response to the industry concern, split pins were replaced at RNP during the 1990 RO-13. A second replacement is currently projected for Spring 2010 RO-26 utilizing improved materials that support the proactive management of aging in reactor internals components. Detailed descriptions of all facets of the replacement are retained in the plant records.

4.3 INDUSTRY PROGRAMS

4.3.1 WCAP-14577, Aging Management for Reactor Internals

The Westinghouse Owner's Group (WOG, now PWROG) topical report WCAP-14577 [27] contains a technical evaluation of aging degradation mechanisms and aging effects for Westinghouse RVI components. The WOG sent the report to the NRC staff to demonstrate that WOG member plant owners that subscribed to the WCAP could adequately manage effects of aging on RVI during the period of extended operation, using approved aging management methodologies of the WCAP to develop plant-specific aging management programs.

The aging management review for the RNP internals, documented in [5, 6] was completed in accordance with the requirements of WCAP-14577 [27].

4.3.2 MRP-227, Reactor Internals Inspection and Evaluation Guidelines

MRP-227, as discussed in Section 2, was developed by a team of industry experts including utility representatives, NSSS vendors, independent consultants, and international committee representatives who reviewed available data and industry experience on materials aging. The objective of the group was to develop a consistent, systematic approach for identifying and prioritizing inspection and evaluation requirements for reactor internals. The following subsections briefly describe the industry process.

4.3.2.1 MRP-227 RVI Component Categorizations

MRP-227 used a screening and ranking process to aid in the identification of required inspections for specific RVI components. MRP-227 credited existing component inspections, when they were deemed adequate, as a result of detailed expert panel assessments conducted in conjunction with the development of the industry document. Through the elements of the process, the reactor internals for all currently licensed and operating PWR designs in the United States were evaluated in the MRP program; and appropriate inspection, evaluation, and implementation requirements for reactor internals were defined.

Based on the completed evaluations, the RVI components are categorized within MRP-227 as "Primary" components, "Expansion" components, "Existing Programs" components, or "No Additional Measures" components, as described as follows:

- <u>Primary</u>

  Those PWR internals that are highly susceptible to the effects of at least one of the eight aging mechanisms were placed in the Primary group. The aging management requirements that are needed to ensure functionality of Primary components are described in the I&E guidelines. The Primary group also includes components that have shown a degree of tolerance to a specific aging degradation effect, but for which no highly susceptible component exists or for which no highly susceptible component is accessible.

- <u>Expansion</u>

  Those PWR internals that are highly or moderately susceptible to the effects of at least one of the eight aging mechanisms, but for which functionality assessment has shown a degree of tolerance to those effects, were placed in the Expansion group. The schedule for implementation of aging management requirements for Expansion components depends on the findings from the examinations of the Primary components at individual plants.

- Existing Programs

  Those PWR internals that are susceptible to the effects of at least one of the eight aging mechanisms and for which generic and plant-specific existing AMP elements are capable of managing those effects, were placed in the Existing Programs group.

- No Additional Measures Programs

  Those PWR internals for which the effects of all eight aging mechanisms are below the screening criteria were placed in the No Additional Measures group. Additional components were placed in the No Additional Measures group as a result of a failure mode, effects, and criticality analysis (FMECA) and the functionality assessment. No further action is required by these guidelines for managing the aging of the No Additional Measures components.

The categorization and analysis used in the development of MRP-227 are not intended to supersede any ASME B&PV Code Section XI requirements. Any components that are classified as core support structures, as defined in ASME B&PV Code Section XI IWB 2500, Category B-N-3, have requirements that remain in effect and may only be altered as allowed by 10 CFR 50.55a.

4.3.2.2 NEI 03-08 Guidance Within MRP-227

The industry program requirements of MRP-227 are classified in accordance with the requirements of the NEI 03-08 protocols. The MRP-227 guideline includes Mandatory, Needed, and Good Practice elements as follows:

- **Mandatory**

There is one Mandatory element:

> *Each commercial U.S. PWR unit shall develop and document a PWR reactor internals aging management program within 36 months following issuance of MRP-227, Rev. 0.*
>
> RNP Applicability: MRP-227 was officially issued by the industry in December 2008. An AMP must therefore be developed by December 2011. Progress Energy is actively developing this AMP for RNP to meet its license renewal commitment that pre-dates the implementation date contained in MRP-227.

- **Needed**

There are three Needed elements:

1. *Each commercial U.S. PWR unit shall implement MRP-227, Tables 4-1 through 4-9 and Tables 5-1 through 5-3 for the applicable design within 24 months following issuance of MRP-227-A.*

   RNP Applicability: MRP-227 augmented inspections have been appropriately incorporated into the RNP ISI for the license renewal period. The applicable Westinghouse tables contained in MRP-227 Table 4.3 (Primary), Table 4.6 (Expansion), and Table 4.9 (Existing) and are attached herein as Appendix C Tables C-1, C-2, and C-3, respectively.

2. *Examinations specified in the MRP-227 guidelines shall be conducted in accordance with Inspection Standard MRP-228.*

   RNP Applicability: Inspection standards will be in accordance with the requirements of MRP-228 [28] when it is issued. These inspection standards will be used for augmented inspection at RNP as applicable where required by MRP-227 directives.

3. *Examination results that do not meet the examination acceptance criteria defined in Section 5 of the MRP-227 guidelines shall be recorded and entered in the plant corrective action program and dispositioned.*

   RNP Applicability: RNP will comply with this requirement.

- **Good Practice**

There is one Good Practice element:

Each commercial U.S. PWR unit should provide a summary report of all inspections and monitoring, items requiring evaluation, and new repairs to the MRP Program Manager within 120 days of the completion of an outage during which PWR internals are examined. The MRP template should be used for the report.

RNP Applicability: As discussed in Section 4, Progress Energy will participate in future industry efforts and will adhere to industry directives for reporting, response, and follow-up.

4.3.2.3 MRP-227 AMP Development Guidance

It should be noted that MRP-227, Appendix A also includes a description of the attributes that make up an acceptable AMP. These attributes are similar to the previously discussed attributes of the GALL Report and are consistent with the RNP Aging Management Review process. Evaluation of the RNP RVI AMP against GALL attribute elements is provided in Section 5 of this program plan.

As part of License Renewal, RNP agreed to participate in industry activities associated with the development of the standard Industry Guideline for Inspection and Evaluation of Reactor Internals. The industry efforts have defined the required inspections and examination techniques for those components critical to aging management of RVI. The results of the industry recommended inspections, as published in MRP-227, serve as the basis for identifying any augmented inspections that are required to complete the RNP RVI AMP.

The MRP-227 guideline has been submitted to the NRC with the ultimate goal of producing an SER and approval. Discussions between utilities and the NRC, however, have indicated that the utility program cannot be based solely on the MRP-227 work, at this time. Therefore, the RNP RVI AMP has been rooted in meeting the GALL attributes and the previous approved work of the WOG (PWROG) WCAP-14577. However, the RNP RVI AMP also captures the results of additional evaluations, inspection recommendations, and forward-looking concepts of the MRP-227 work summarized in the tables of Appendix C. It should be noted that the contents of MRP-227 fully align with the information and NRC-approved directives contained in WCAP-14577.

4.3.2.4 MRP-227 Applicability to RNP

The applicability of MRP-227 to RNP requires compliance with the following MRP-227 assumptions:

- *Operation of 30 years or less with high-leakage core loading patterns (fresh fuel assemblies loaded in peripheral locations) followed by implementation of a low-leakage fuel management strategy for the remaining 30 years of operation.*

  RNP Applicability: RNP fuel management program changed from a high- to a low-leakage core loading pattern prior to 30 years of operation.

- *Base load operation, i.e., typically operates at fixed power levels and does not usually vary power on a calendar or load demand schedule.*

  RNP Applicability: RNP operates as a base load unit.

- *No design changes beyond those identified in general industry guidance or recommended by the original vendors.*

  RNP Applicability: MRP-227 states that the recommendations are applicable to all U.S. PWR operating plants as of May 2007 for the three designs considered. RNP has not made any modifications to reactor internals components since May 2007. RNP plans to replace split pins with an upgraded material in 2010. The modification will have no impact on the applicability of MRP-227 and is an example of the RNP proactive approach to managing aging reactor internals.

Based on the RNP applicability, as stated, the MRP-227 work is representative for Robinson Nuclear Plant.

4.3.3 Ongoing Industry Programs

The U.S. industry, through both the EPRI/MRP and the PWROG, continues to sponsor activities related to RVI aging management, including planned development of a standard NRC submittal template, development of a plant-specific implementation program template for currently licensed U.S. PWR plants, and development of acceptance criteria and inspection disposition processes. Progress Energy will maintain cognizance of industry activities related to PWR internals inspection and aging management; and will address/implement industry guidance, stemming from those activities, as appropriate under NEI 03-08 practices.

4.4 SUMMARY

It should be noted that the Progress Energy RNP, the MRP, and the PWROG approaches to aging management are based on the GALL approach to aging management strategies. This approach includes a determination of which reactor internals passive components are most susceptible to the aging mechanisms of concern and then determination of the proper inspection or mitigating program that provides reasonable assurance that the component will continue to perform its intended function through the period of extended operation. The GALL-based approach was used at RNP for the initial basis of the LRA that resulted in the NRC SER in NUREG-1785.

The approach used to develop RNP AMPs is fully compliant with regulatory directives and approved documents. The additional evaluations and analysis completed by the MRP industry group have provided clarification to the level of inspection quality needed to determine the proper examination method and frequencies. The tables provided in MRP-227 and included as Appendix C of this AMP provide the level of examination required for each of the components evaluated.

It is the RNP position that use of the AMR produced by the LRA methodology, combined with any additional augmented inspections required by the MRP-227 industry tables provided in Appendix C, provides reasonable assurance that the reactor internals passive components will continue to perform their intended functions through the period of extended operation.

5 H. B. ROBINSON REACTOR INTERNALS AGING MANAGEMENT PROGRAM ATTRIBUTES

The RNP RVI AMP is credited for aging management of RVI components for the following eight aging degradation mechanisms and their associated effects:

- Stress corrosion cracking
- Irradiation-assisted stress corrosion cracking
- Wear
- Fatigue
- Thermal aging embrittlement
- Irradiation embrittlement
- Void swelling and irradiation growth
- Thermal and irradiation-enhanced stress relaxation or irradiation-enhanced creep

The attributes of the RNP Reactor Internals AMP and compliance with NUREG-1801 (GALL Report), Section XI.M16, "PWR Vessel Internals" [4] are described in this section. The GALL identifies 10 attributes for successful component aging management. The framework for assessing the effectiveness of the projected program is established by the use of the 10 elements of the GALL.

RNP fully utilized the GALL process contained in NUREG-1801 [4] in performing the aging management review of the reactor internals in the license renewal process. However, RNP made a commitment (see NUREG-1785 [2]) to incorporate the following: (1) RNP will continue to participate in industry programs to investigate aging effects and determine the appropriate AMP activities to address baffle and former assembly issues, and to address change in dimensions due to void swelling, (2) as WOG and EPRI MRP research projects are completed, RNP will evaluate the results and factor them into the PWR Vessel Internals Program as appropriate, and (3) RNP will implement an augmented inspection during the license renewal term. Augmented inspections, based on required program enhancements resulting from industry programs, will become part of the ASME B&PV Code, Section XI program.

This AMP is consistent with that process and includes consideration of the augmented inspections identified in MRP-227 and fully meets the requirements of the commitment. Specific details of the RNP Reactor Internals AMP are summarized in the following subsections.

5.1 GALL ELEMENT 1: SCOPE OF PROGRAM

GALL Report AMP Element Description

"The program is focused on managing the effects of crack initiation and growth due to stress corrosion cracking (SCC) or irradiation assisted stress corrosion cracking (IASCC), and loss of fracture toughness due to neutron irradiation embrittlement or void swelling. The program contains preventive measures to mitigate SCC or IASCC; ISI to monitor the effects of cracking on the intended function of the components; and repair and/or replacement as needed to maintain the ability to perform the intended function. Loss of fracture toughness is of consequence only if cracks exist. Cracking is expected to initiate at the surface and is detectable by augmented inspection. The program provides guidelines to assure safety function integrity of the subject

safety related reactor pressure vessel internal components, both non-bolted and bolted components.

The program consists of the following elements: (a) identify the most susceptible or limiting items, (b) develop appropriate inspection techniques to permit detection and characterizing of the feature (cracks) of interest and demonstrate the effectiveness of the proposed technique, and (c) implement the inspection during the license renewal term. For example, appropriate inspection techniques may include enhancing visual VT-1 examinations for non-bolted components and demonstrated acceptable inspection methods for bolted components." [4]

RNP Program Scope

The RNP RVI consist of two basic assemblies: (1) an upper internals assembly that is removed during each refueling operation to obtain access to the reactor core, and (2) a lower internals assembly that can be removed, if desired, following a complete core unload. Additional RVI details are provided in Sections 3.9.5 and 4.1 of the RNP UFSAR.

The RNP RVI subcomponents that required aging management review are indicated in the previously submitted Table 2.3-1 of the RNP Application for Renewal Operating License [38]. The portion of this table associated with the internals is included as part of the tables in Appendix B. The table lists the subcomponents of the RVI that required aging management review along with each subcomponent passive function(s) and reference(s) to the corresponding AMR table(s) in Section 3 of the RNP License Renewal Application.

The RNP Reactor Internals Aging Management Review was conducted and documented in the RNP Aging Management Review Calculations [5, 6]. The table summarizing the results of that review is also included in the tables of Appendix B. The tables identify those aging effects that require management for those components requiring AMR. A column in the tables lists the program/activity that is credited to address the component and aging effect during the period of extended operation. The NRC has reviewed and approved the aging management strategy presented in the Appendix B tables as documented in the SER on license renewal [2].

The results of the industry research provided by MRP-227, summarized in the tables of Appendix C, provide the basis for the required augmented inspections, inspection techniques to permit detection and characterizing of the aging effects (cracks, loss of material, loss of preload, etc.) of interest, prescribed frequency of inspection, and examination acceptance criteria. The RNP RVI AMP scope is based on previously established and approved GALL Report approaches through application of the WCAP-14577 methodologies to determine those components that require aging management. Likewise, the additional information provided in the industry MRP-227 (results of Appendix C) is rooted in the GALL methodology and provides a basis for augmented inspections that were required to complete this RNP RVI AMP by providing the inspection method, frequency of inspection, and examination acceptance criteria.

Conclusion

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16 and Commitment 33 in the RNP SER.

5.2 GALL ELEMENT 2: PREVENTIVE ACTIONS

GALL Report AMP Element Description

> *"The requirements of ASME Section XI, Subsection IWB, provide guidance on detection, but do not provide guidance on methods to mitigate cracking. Maintaining high water purity reduces susceptibility to cracking due to SCC. Reactor coolant water chemistry is monitored and maintained in accordance with the EPRI guidelines in TR-1014986. The program description and evaluation and technical basis of monitoring and maintaining reactor water chemistry are presented in Chapter XI.M2, Water Chemistry."*[4]

RNP Preventive Action

The RNP reactor internals AMP includes the following existing program that complies with the requirements of this element. A description and applicability to the RNP reactor internals AMP is provided in the following subsection.

Primary Water Chemistry Program

To mitigate aging effects on component surfaces that are exposed to water as process fluid, chemistry programs are used to control water chemistry for impurities (e.g., dissolved oxygen, chloride, fluoride, and sulfate) that accelerate corrosion. This program relies on monitoring and control of water chemistry to keep peak levels of various contaminants below the system-specific limits. The RNP PWR Primary Water Chemistry Program [15, 16] is based on the current, approved revisions of EPRI PWR Primary Water Chemistry Guidelines.

This program is consistent with the corresponding program described in the GALL Report [23].

The limits of known detrimental contaminants imposed by the chemistry monitoring program are consistent with the EPRI PWR Primary Water Chemistry Guidelines [20].

Conclusion

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16 and Commitment 33 in the RNP SER.

5.3 GALL ELEMENT 3: PARAMETERS MONITORED OR INSPECTED

GALL Report AMP Element Description

> *"The program monitors the effects of cracking on the intended function of the component by detection and sizing of cracks by augmentation of the inservice inspection requirements in accordance with the requirements of the ASME Code Section XI, Table IWB 2500-1."* [4]

RNP Parameters Monitored or Inspected

The RNP AMP monitors, inspects, and/or tests for the effects of the eight aging degradation mechanisms on the intended function of the RNP PWR internals components through inspection and condition monitoring activities in accordance with the augmented requirements defined under industry directives as contained in MRP-227 and ASME Section XI.

For license renewal, the ASME Section XI Program consists of periodic volumetric, surface, and/or visual examination of components for assessment, signs of degradation, and corrective actions. This program is consistent with the corresponding program described in the GALL Report [23].

Appendices B and C of this AMP provide a detailed listing of the components and subcomponents and the parameters monitored, inspected, and/or tested.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16 and Commitment 33 in the RNP SER.

5.4 GALL ELEMENT 4: DETECTION OF AGING EFFECTS

GALL Report AMP Element Description

> *"The extent and schedule of the inspection and test techniques prescribed by the aging management program are designed to maintain structural integrity and ensure that aging effects will be discovered and repaired before the loss of intended function. Inspections can reveal crack initiation and growth. Vessel internal components that are inspected in accordance with the requirements of ASME Section XI, Subsection IWB examination category B-N-3 for all accessible surfaces of reactor core support structure that can be removed from the vessel. The ASME Section XI inspection specifies visual VT-3 examination to determine the general mechanical and structural condition of the component supports by (a) verifying parameters such as clearances, settings, and physical displacements, and (b) detecting discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion.*
>
> *However, visual VT-3 examination is to be augmented to detect tight or fine cracks. Also, historically the VT-3 examination have not identified bolt cracking because cracking occurs at the juncture of the bolt head and shank, which is not accessible for visual inspection. Creviced*

and other inaccessible regions are difficult to inspect visually. This AMP recommends more stringent inspections such as enhanced visual VT-1 examinations or ultrasonic methods of volumetric inspection, for certain selected components and locations.

The inspection technique is capable of detecting the critical flaw size with adequate margin. The critical flaw size is determined based on the service loading condition and service-degraded material properties. For non-bolted components, augmented ISI may include enhancement of the visual VT-1 examination of Section XI IWA-2210. A description of such an enhanced visual VT-1 examination should include the ability to achieve a 0.0005-in. resolution, with the conditions (e.g., lighting and surface cleanliness) of the inservice examination bounded by those used to demonstrate the resolution of the inspection technique. For bolted components, augmented ISI is to include other demonstrated acceptable inspection methods to detect cracks between the bolt head and the shank. Alternatively, the applicant may perform a component-specific evaluation, including a mechanical loading assessment to determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. If the loading is compressive or low enough (<5 ksi) to preclude fracture, then supplemental inspection of the component is not required. Failure to meet this criterion requires continued use of the augmented inspection methods." [4]

RNP Detection of Aging Effects

Detection of indications that are required by the ASME Section XI ISI Program is well established and field-proven through the application of the Section XI ISI Program. Those augmented inspections that are taken from the MRP-227 recommendations will be applied through use of the MRP-228 Inspection Standard.

Inspection can be used to detect physical effects of degradation including cracking, fracture, wear, and distortion. The choice of an inspection technique depends on the nature and extent of the expected damage. The recommendations supporting aging management for the reactor internals, as contained in this report, are built around three basic inspection techniques: (1) visual, (2) ultrasonic, and (3) physical measurement. Three different visual techniques are include VT-3, VT-1, and EVT-1. The assumptions and process used to select the appropriate inspection technique are described in the following subsections. Inspection standards developed by the industry for the application of these techniques for augmented reactor internals inspections are documented in MRP-228.

<u>VT-1 Visual Examinations</u>

The acceptance criteria for visual examinations conducted under categories B-N-2 (welded core support structures and interior attachments to reactor vessels) and B-N-3 (removable core support structures) are defined in IWB-3520 [42]. VT-1 visual examination is intended to identify crack-like surface flaws. Unacceptable conditions for a VT-1 examination are:

- Crack-like surface flaws on the welds joining the attachment to the vessel wall that exceed the allowable linear flaw standards of IWB-3510

- Structural degradation of attachment welds such that the original cross-sectional area is reduced by more than 10 percent

These requirements are defined to ensure the integrity of attachment welds on the ferritic pressure vessel. Although the IWB-3520 criteria do not directly apply to austenitic stainless steel internals, the clear intent is to ensure that the structure will meet minimum flaw tolerance fracture requirements. In the MRP-227 recommendations, VT-1 examinations have been identified for components requiring close visual examinations with some estimate of the scale of deformation or wear. In MRP-227 note that VT-1 has only been selected to detect distortion as evidenced by small gaps between the upper-to-lower mating surfaces of CE-welded core shrouds assembled in two vertical sections. Therefore, no additional VT-1 inspections over and above those required by ASME Section XI ISI have been specified.

EVT-1 Enhanced Visual Examination for the Detection of Surface Breaking Flaws

In the augmented inspections detailed in the MRP-227 for reactor internals, the EVT-1 enhanced visual examination has been identified for inspection of components where surface-breaking flaws are a potential concern. Any visual inspection for cracking requires a reasonable expectation that the flaw length and crack mouth opening displacement meet the resolution requirements of the observation technique. The EVT-1 specification augments the VT-1 requirements to provide more rigorous inspection standards for stress corrosion cracking and has been demonstrated for similar inspections in boiling water reactor (BWR) internals. Enhanced visual examination (i.e., EVT-1) is also conducted in accordance with the requirements described for visual examination (i.e., VT-1) with additional requirements (such as camera scanning speed) currently being developed by the industry. Any recommendation for EVT-1 inspection will require additional analysis to establish flaw-tolerance criteria. The industry is currently developing a consensus approach for acceptance criteria methodologies to support plant-specific augmented examinations. Progress Energy has been an active participant in these initiatives and will follow the industry directive. These acceptance criteria methodologies may be determined either generically or on a plant-specific basis because both loads and component dimensions may vary from plant to plant within a typical PWR design.

VT-3 Examination for General Condition Monitoring

In the augmented inspections detailed in the MRP-227 for reactor internals, the VT-3 visual examination has been identified for inspection of components where general condition monitoring is required. The VT-3 examination is intended to identify individual components with significant levels of existing degradation. As the VT-3 examination is not intended to detect the early stages of component cracking or other incipient degradation effects, it should not be used when failure of an individual component could threaten either plant safety or operational stability. The VT-3 examination may be appropriate for inspecting highly redundant components (such as baffle-edge bolts), where a single failure does not compromise the function or integrity of the critical assembly.

The acceptance criteria for visual examinations conducted under categories B-N-2 (welded core support structures and interior attachments to reactor vessels) and B-N-3 (removable core support structures) are defined in IWB-3520. These criteria are designed to provide general guidelines. The unacceptable conditions for a VT-3 examination are:

- Structural distortion or displacement of parts to the extent that component function may be impaired;
- Loose, missing, cracked, or fractured parts, bolting, or fasteners;
- Foreign materials or accumulation of corrosion products that could interfere with control rod motion or could result in blockage of coolant flow through fuel;
- Corrosion or erosion that reduces the nominal section thickness by more than 5 percent;
- Wear of mating surfaces that may lead to loss of function;
- Structural degradation of interior attachments such that the original cross-sectional area is reduced more than 5 percent.

The VT-3 examination is intended for use in situations where the degradation is readily observable. It is meant to provide an indication of condition, and quantitative acceptance criteria are not generally required. In any particular recommendation for VT-3 visual examination, it should be possible to identify the specific conditions of concern. For instance, the unacceptable conditions for wear indicate wear that might lead to loss of function. Guidelines for wear in a critical-alignment component may be very different from the guidelines for wear in a large structural component.

Ultrasonic Testing

Volumetric examinations in the form of ultrasonic testing (UT) techniques can be used to identify and determine the length and depth of a crack in a component. Although access to the surface of the component is required to apply the ultrasonic signals, the flaw may exist in the bulk of the material. In this proposed strategy, UT inspections have been recommended exclusively for detection of flaws in bolts. For the bolt inspections, any bolt with a detected flaw should be assumed to have failed. The size of the flaw in the bolt is not critical because crack growth rates are generally high, and it is assumed that the observed flaw will result in failure prior to the next inspection opportunity. It has generally been observed through examination performance demonstrations that UT can reliably (90 percent or greater reliability) detect flaws that reduce the cross-sectional area of a bolt by 35 percent.

Failure of a single bolt does not compromise the function of the entire assembly. Bolting systems in the reactor internals are highly redundant. For any system of bolts, it is possible to demonstrate multiple minimum acceptable bolting patterns. The evaluation program must demonstrate that the remaining bolts meet the requirements for a minimum bolting pattern for continued operation. The evaluation procedures must also demonstrate that the pattern of remaining bolts contains sufficient margin such that continuation of the bolt failure rate will not result in failure of the system to meet the requirements for minimum acceptable bolting pattern before the next inspection.

Establishment of the minimum acceptable bolting pattern for any system of bolts requires analysis to demonstrate that the system will maintain reliability and integrity in continuing to perform the intended function of the component. This analysis is highly plant-specific. Therefore, any recommendation for UT inspection of bolts assumes that the plant owner will work with the designer to establish minimum acceptable bolting patterns prior to the inspection to support continued operation. For Westinghouse-designed plants, minimum acceptable bolting patterns for baffle-former and barrel-former bolts are available through the PWROG. Progress Energy has been a full participant in the development of the PWROG documents and has full access and use.

Physical Measurement Examination

Continued functionality can be confirmed by physical measurements to evaluate the impact caused by various degradation mechanisms such as wear or loss of functionality as a result of loss of preload or material deformation. For RNP, direct physical measurements are required only for the hold-down spring.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16 and Commitment 33 in the RNP SER.

5.5 GALL ELEMENT 5: MONITORING AND TRENDING

GALL Report AMP Element Description

> *"Inspection schedules in accordance with IWB-2400, assessment of susceptible or limiting components or locations, and reliable examination methods provide timely detection of cracks. The scope of examination expansion and reinspection beyond the baseline inspection are required if flaws are detected."* [4]

RNP Monitoring and Trending

Operating experience with PWR reactor internals has been generally proactive. Flux thimble wear and control rod guide tube split pin cracking issues were identified by the industry and continue to be actively managed. The extremely low frequency of failure in reactor internals makes monitoring and trending based on operating experience somewhat impractical. The majority of the materials aging degradation models used to develop the MRP-227 Guidelines are based on test data from reactor internals components removed from service. The data is used to identify trends in materials degradation and forecast potential component degradation. The industry continues to share both material test data and operating experience through the auspices of the MRP and PWROG. Progress Energy has in the past and will continue to maintain cognizance of industry activities and shared information related to PWR internals inspection and aging management.

Inspections credited in Appendix B are based on utilizing the RNP 10-year ISI program and the augmented inspections derived from the industry program contained in Appendix C. These inspections, where practical, are scheduled to be conducted in conjunction with typical 10-year ISI examinations.

Appendix C, Tables C-1, C-2, and C-3 identify the augmented primary and expansion inspection and monitoring recommendations, and the existing programs credited for inspection and aging management. As discussed in MRP-227, inspection of the "Primary" components provides reasonable assurance for demonstrating component current capacity to perform the intended functions.

Reporting requirements are included as part of the MRP-227 guidelines. Consistent reporting of inspection results across all PWR designs will enable the industry to monitor reactor internals degradation on an ongoing industry basis as the period of extended operation moves forward. Reporting of examination results will allow the industry to monitor and trend results and take appropriate preemptive action through update of the MRP guidelines.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16 and Commitment 33 in the RNP SER.

5.6 GALL ELEMENT 6: ACCEPTANCE CRITERIA

GALL Report AMP Element Description

> *"Any indication or relevant condition of degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3500."* [4]

RNP Acceptance Criteria

Those recordable indications that are the result of inspections required by the existing RNP ISI program scope are evaluated in accordance with the applicable requirements of the ASME Code through the existing Corrective Action Program [29].

Inspection acceptance and expansion criteria are provided in Appendix C, Table C-4. These criteria will be reviewed periodically as the industry continues to develop and refine the information and will be included in updates to RNP procedures to enable the examiner to identify examination acceptance criteria considering state-of-the-art information and techniques.

Augmented inspections, as defined by the MRP-227 requirements, that result in recordable relevant conditions will be entered into the plant Corrective Action Program and addressed by appropriate actions that may include enhanced inspection, repair, replacement, mitigation actions, or analytical evaluations. An example of an analytical evaluation is using a minimum bolting WCAP approach such as those commonly used to support continued component or assembly functionality. Additional analysis to establish acceptable bolting pattern evaluation criteria for the baffle-former bolt assembly, as contained in various industry documents [39], is also considered in determining the acceptance of inspection results to support continued component or assembly functionality. The industry, through various cooperative efforts, is working to construct a consensus set of tools in line with accepted and proven methodologies to support this element. Additional analysis to establish Appendix C expansion component evaluation criteria is being performed through the efforts of the PWROG. Status is monitored through direct

Progress Energy cognizance of industry (including PWROG) activities related to PWR internals inspection and aging management.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16 and Commitment 33 in the RNP SER.

5.7 GALL ELEMENT 7: CORRECTIVE ACTIONS

GALL Report AMP Element Description

> *"Repair and replacement procedures are equivalent to those requirements in ASME Section XI. Repair is in conformance with IWB-4000 and replacement occurs according to IWB-7000. As discussed in the regulatory documents integral with the GALL report, the staff considered the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions."* [4]

RNP Corrective Action

The existing RNP procedure for Inservice Repair and Replacement [43] will be credited for this element. This procedure establishes the RNP repair and replacement requirements of ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." These requirements include the identification of a repair cycle, repair plan, and verification of acceptability for replacements. RNP committed that corrective actions for augmented inspections will be developed using repair and replacement procedures equivalent to those requirements in ASME B&PV Code, Section XI.

Conclusion

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16 and Commitment 33 in the RNP SER.

5.8 GALL ELEMENT 8: CONFIRMATION PROCESS

GALL Report AMP Element Description

> *"Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls."* [4]

RNP Confirmation Process

RNP has an established 10 CFR Part 50, Appendix B, Program [34, 35, 36] that addresses the elements of corrective actions, confirmation process, and administrative controls. The RNP Program includes non-safety-related structures, systems, and components. Quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16 and Commitment 33 in the RNP SER.

5.9 GALL ELEMENT 9: ADMINISTRATIVE CONTROLS

GALL Report AMP Element Description

See item 8 (Section 5.8).

RNP Administrative Controls

See evaluation in Section 5.8.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16 and Commitment 33 in the RNP SER.

5.10 GALL ELEMENT 10: OPERATING EXPERIENCE

GALL Report AMP Element Description

> *"Because the ASME Code is a consensus document that has been widely used over a long period, it has been shown to be generally effective in managing aging effects in Class 1, 2, or 3 components and their integral attachments in light-water cooled power plants.*
>
> *In PWRs, stainless steel components have generally not been found to be affected by SCC because of low dissolved oxygen levels and control of primary water chemistry. However, the potential for SCC exists due to inadvertent introduction of contaminants into the primary coolant system from unacceptable levels of contaminants in the boric acid; introduction through the free surface of the spent fuel pool, which can be a natural collector of airborne contaminants (NRC IN 84-18); introduction of relatively high levels of oxygen during shutdown, or from aggressive chemistries that may develop in creviced regions. Cracking has occurred in SS baffle former bolts in a number of foreign plants (NRC IN 98-11) and has now been observed in plants in the United States."*[4]

The ASME Code is a consensus document that has been widely used over a long period of time; it has been shown to be generally effective in managing aging effects in Class 1, 2, or 3 components and their integral attachments in light-water-cooled power plants.

RNP Operating Experience

Extensive industry and RNP operating experience has been reviewed during the development of the Reactor Vessel Internals Aging Management Program. The experience reviewed includes NRC Information Notices 84-18, "Stress Corrosion Cracking in PWR Systems" and 98-11, "Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants." Most of the industry operating experience reviewed has involved cracking of austenitic stainless steel baffle-former bolts, or SCC of high-strength internals bolting. SCC of control rod guide tube split pins has also been reported.

Early plant operating experience related to hot functional testing and reactor internals is documented in plant historical records. Inspections performed as part of the 10-year ISI program have been conducted as designated by existing commitments and would be expected to discover overall general internals structure degradation. To date, very little degradation has been observed industry-wide.

Industry operating experience is routinely reviewed by Progress Energy engineers using INPO Operating Experience (OE), the Nuclear Network, and other information sources as directed under the applicable procedure [30], for the determination of additional actions and lessons learned. These insights, as applicable, can be incorporated into the plant systems quarterly health reports and further evaluated for incorporation into plant programs.

A review of industry and plant-specific experience with reactor vessel internals reveals that the U.S. industry, including Progress Energy and RNP, has responded proactively to industry issues relative to reactor internals degradation. Two examples that demonstrate this proactive response are the replacement of control rod guide tube split pins in 1990 and participation by Progress Energy in the augmented examinations to be performed by the PWROG on control rod guide tube guide cards. These are briefly described in the following paragraphs.

- <u>RNP Control Rod Guide Tubes Split Pins</u>

The control rod guide tube split pins were replaced at RNP during the 1990 refueling outage. The new pins were considered a replacement in kind and an improvement over the previous pins. Ongoing industry experience has shown continued degradation of this component and, in response, RNP is planning to perform a second split pin replacement activity in spring 2010 RO-26. The replacement will include a material upgrade from X-750 to 316SS in support of managing aging in the component.

- <u>RNP Participation as a Representative Pilot Plant in the PWROG Control Rod Guide Cards Inspection Program</u>

The PWROG is currently conducting upper internals control rod guide tube card wear measurements on a sample of guide tubes from selected representative pilot plants to approximate the remaining life of the guide tube guide cards. This is a proactive effort by the U.S. industry to establish criteria for inspection and gather data to support aging management of the component. RNP proactively inspected guide cards in 1990 RO-13, and is now a key participant as one of the representative pilot inspection plants. The RNP inspection in support of the industry program is targeted for spring 2010 RO-26. Inspection outcomes will be evaluated to ensure compliance with MRP-227 specifications.

A key element of the MRP-227 Guideline is the reporting of age-related degradation of reactor vessel components. Progress Energy, through its participation in PWROG and EPRI-MRP activities, will continue to benefit from the reporting of inspection information and will share its own operating experience with the industry through those groups or INPO, as appropriate.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16 and Commitment 33 in the RNP SER.

6 DEMONSTRATION

Robinson Nuclear Plant has demonstrated a long-term commitment to aging management of reactor internals. This AMP is based on an established history of programs to identify and monitor potential aging degradation in the reactor internals. Programs and activities undertaken in the course of fulfilling that commitment include:

- The examinations required by ASME Section XI for the RNP reactor vessel internals have been performed during each 10-year interval since plant operations commenced.
- As documented in RNP operational procedures, reports are continuously reviewed by RNP personnel for applicable issues that indicate operating procedures or programs require updates based on new OE.
- Review of Nuclear Oversight Section (NOS) audit reports, NRC inspection reports, and INPO evaluations indicate no unacceptable issues related to reactor vessel internals inspections.
- The Primary Water Chemistry Program at RNP has been effective in maintaining oxygen, halogens, and sulfate at levels sufficiently low to prevent SCC of the reactor vessel internals
- Replacement control rod guide tube split pins for RNP in 1990 were fabricated from more resistant Alloy X-750 materials, with modified geometry and heat treatment to increase resistance to SCC (versus original pins). The new Type 316 stainless steel split pins for replacement in spring 2010 will provide additional resistance to PWSCC.
- Progress Energy has actively participated in past and ongoing EPRI and PWROG RVI activities. Progress Energy will continue to maintain cognizance of industry activities related to PWR internals inspection and aging management; and will address/implement industry guidance, stemming from those activities, as appropriate under NEI 03-08 practices.

This AMP fulfills the approved license renewal methodology requirement to identify the most susceptible components and to inspect those components with an indication detection level commensurate with the expected degradation mechanism indication. Augmented inspections, derived from the information contained in MRP-227, the industry I&E Guidelines, have been utilized in this AMP to build on existing plant programs. This approach is expected to encourage detection of a degradation mechanism at its first appearance consistent with the ASME approach to inspections. This approach provides reasonable assurance that the internals components will continue to perform their intended function through the period of extended operation.

Typical ASME Section XI examinations identified in the AMP for the period of extended operation are currently scheduled to be performed at RNP beginning in Fall 2011 RO-27. The augmented inspections discussed in compliance with MRP-227 requirements will be integrated in the inspection procedures used to perform the ASME Section XI 10-year ISI examinations. Integration of the required inspections will be tracked to completion. As discussed, the industry MRP-227 guidelines also provide for updates as experience is gained through inspection results. This feedback loop will enable updates based on actual inspection experience.

The augmented inspections described in this document, as summarized in Appendix C, combined with the ASME Section XI ISI program inspections, existing RNP programs, and use of OERs, provide reasonable assurance that the reactor internals will continue to perform their intended functions through the period of extended operation.

7 PROGRAM ENHANCEMENT AND IMPLEMENTATION SCHEDULE

The requirements of MRP-227 are based on an 18-month refueling cycle and consider both EFPY and cumulative operation. The information contained in Table 7-1 is based on this information and includes a description of the currently projected scope of inspection pertaining to the reactor internals AMP. Should a change occur in plant operational practices or operating experience result in changes to the projections, appropriate updates will be performed on affected plant documentation in accordance with approved procedures.

Table 7-1 Aging Management Program Enhancement and Inspection Implementation Summary

Refueling Outage	Project Month/Year	Estimated EFPY	AMP-Related Scope	Inspection Method and Criteria	Comments
27	Fall 2011	30.4	ASME Code Section XI		Third period of fourth internal (RVI inspections)
28	Spring 2013	31.9	MRP-227 augmented inspections for control rod guide tube guide cards, control rod guide tube lower flanges, upper core barrel flange weld, and thermal shield flexures	MRP-227 inspections in accordance with MRP-228 specifications	Not applicable
29	Fall 2014	33.4	MRP-227 augmented inspections for internals hold-down spring inspection	MRP-227 inspections in accordance with MRP-228 specifications Flux thimble tube eddy current inspections for loss of material (wear)	May exercise option to do a preemptive replacement of the hold-down spring instead of MRP-227 inspections Flux thimble tube existing program is identified in MRP-227 as a plant-specific inspection
30	Spring 2016	34.9	MRP-227 augmented inspections for the baffle-former bolts	MRP-227 inspections in accordance with MRP-228 specifications	Not applicable
31	Fall 2017	36.4	Not applicable	Not applicable	Not applicable
32	Spring 2019	37.9	Not applicable	Not applicable	Not applicable

Table 7-1 Aging Management Program Enhancement and Inspection Implementation Summary

Refueling Outage	Project Month/Year	Estimated EFPY	AMP-Related Scope	Inspection Method and Criteria	Comments
33	Fall 2020	39.4	ASME Code Section XI and MRP-227 augmented inspections to include the baffle-edge bolts and the baffle-former assembly	MRP-227 inspections in accordance with MRP-228 specifications	Third period of fifth internal (RVI inspections)
34	Spring 2022	40.9	Not applicable	Not applicable	Not applicable
35	Fall 2023	42.4	Not applicable	Not applicable	Not applicable
36	Spring 2025	43.9	Not applicable	Not applicable	Not applicable
37	Fall 2026	45.4	Not applicable	Not applicable	Not applicable
38	Spring 2028	46.9	Not applicable	Not applicable	Not applicable
39	Fall 2029	48.4	Not applicable	Not applicable	Renewed Operating License expires 7/31/2030

8 IMPLEMENTING DOCUMENTS

As noted within this RNP AMP document, the PWR Vessel Internals Program is a part of the "Progress Energy Reactor Coolant System Material Integrity Management Program" as documented in ADM-NGGC-0112 [7]. The RNP AMP also references the Water Chemistry Program and the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program. MRP-227 augmented examinations (Appendix C), recommended as a result of industry programs, will be included in the existing ASME Section XI program.

RNP documents associated with the existing RNP programs and considered to be implementing documents of the PWR Vessel Internals Program are:

- CP-200, Chemistry Program Implementation
- PLP-025, Inservice Inspection Programs
- TMM-038, Inservice Examination Program

The PWR Vessel Internals AMP relies on the Water Chemistry Program for maintaining high water purity to reduce susceptibility to cracking due to SCC. The Water Chemistry Program was evaluated [2, 23] and found to be consistent with GALL with some exceptions related to augmented inspections expected to be defined through industry programs. Additional procedures may be updated or created as OE for augmented examinations is accumulated.

Based on this information, the updated AMP for RNP RVI provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the CLB for the period of extended operation.

9 REFERENCES

1. 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."
2. NUREG-1785, "Safety Evaluation Report Related to the License Renewal of H. B. Robinson Steam Electric Plant, Unit 2," Docket No. 50-261. Carolina Power & Light Company, March 2004.
3. NUREG-1800, U.S. NRC Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants (SRP-LR), April 2001.
4. NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," April 2001.
5. RNP-L/LR-0354A, Rev. 2, "Nuclear Generation Group Aging Management Review for Reactor Vessel (1005)."
6. RNP-L/LR-0354B, Rev. 1, "CP&L a Progress Energy Company, Aging Management Review for Reactor Internals (1005) for H. B. Robinson – Unit No. 2."
7. ADM-NGGC-0112, Rev. 1, "Reactor Coolant System Material Integrity Management Program."
8. RNP-L/LR-0614, Rev. 2, "Aging Management Program PWR Vessel Internals Program."
9. EGR-NGGC-0504, Rev. 8, "Mechanical System Aging Management Review for License Renewal."
10. EGR-NGGC-0506, Rev. 7, "Civil/Structural Screening and Aging Management Review for License Renewal."
11. Materials Reliability Program: "PWR Internals Inspection and Evaluation Guidelines" (MRP-227-Rev. 0), EPRI, Palo Alto, CA, December 2008. 1016596.
12. NEI 03-08, "Guidelines for the Management of Materials Issues," Nuclear Energy Institute, Washington, DC, December 2008 effective version.
13. PLP-025, Rev. 20, "In-service Inspection Programs."
14. TMM-038, Rev. 13, "Inservice Examination Program."
15. CP-200, Rev. 14, "H. B. Robinson Steam Electric Plant Unit No. 2 Plant Operating Manual Volume 5 Part 3 Chemistry Program Implementation."
16. RNP-L/LR-0600, Rev. 11, "Aging Management Program Water Chemistry Program."
17. RNP-L/LR-0606, Rev. 4, "Aging Management Program ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program."
18. RNP-L/LR-0609, Rev. 1, "Aging Management Program, Flux Thimble Eddy Current Inspection Program."
19. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1995 Edition, 1996 Addenda, American Society of Mechanical Engineers, New York, NY.

20. "Pressurized Water Reactor Primary Water Chemistry Guidelines," Volumes 1 and 2, Revision 6, EPRI, Palo Alto, CA; 2007, 1014986.

21. U.S. NRC Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors," July 26, 1988.

22. EGR-NGGC-0503, Rev. 9, "Mechanical Component Screening for License Renewal."

23. H. B. Robinson UFSAR; Section 3.9.5; Section 4.1 ; Section 18.1.

24. Westinghouse Report WCAP-12202, "Bottom Mounted Instrumentation Double Wall Flux Thimble Tube Combination Thimble Tube Wear Program," February 1989.

25. EC56266, Rev. 11, "Reactor Head & Service Structure Replacement."

26. Engineering Evaluation 90-079, "Control Rode Guide Tube Support Pin Replacement."

27. Westinghouse Report WCAP-14577, Rev. 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals," March 2001.

28. Materials Reliability Program: Inspection Standard for Reactor Internals (MRP-228). EPRI, Palo Alto, CA: 2009. 1016609 (publication pending).

29. CAP-NGGC-0200, Rev. 27, "Nuclear Generation Group Standard Procedure – Corrective Action Program."

30. CAP-NGGC-0202, Rev. 14, "Nuclear Generation Group Standard Procedure – Operating Experience Program."

31. CAP-NGGC-0205, Rev. 9, "Nuclear Generation Group Standard Procedure – Significant Adverse Condition Investigations and Adverse Condition Investigations-increased Rigor,"

32. CAP-NGGC-0206, Rev. 3, "Nuclear Generation Group Standard Procedure – Corrective Action Program Trending and Analysis."

33. Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals Components of Westinghouse and Combustion Engineering PWR Design (MRP-191), EPRI, Palo Alto CA: 2006 1013234.

34. NGGM-PM-0007, Rev. 15, "Quality Assurance Program Manual."

35. PRO-NGGC-0200, Rev. 9, "Procedure Use and Adherence."

36. PRO-NGGC-0204, Rev. 15, "Procedure Review and Approval."

37. Westinghouse Report WCAP-15028, "Guide Tube Cold-Worked 316 Replacement Support Pin Development Program," March 1998.

38. Letter J.W. Moyer, Carolina Power and Light Company, to the U.S. Nuclear Regulatory Commission, Subject: Application for Renewal of Operating License, H.B. Robinson Steam Electric Plant, Unit 2, June 14, 2002.

39. WCAP-15664 "Determination of Acceptable Baffle-Barrel-Bolting for Three-Loop Westinghouse 15x15 Downflow and 17x17 Standard Upflow Domestic Plants," December 2001.

40. EC 50826, Rev. 1, "Cut and Cap Flux Thimbles N-5 and N-12 During RO-21."

41. 5379-04731, Rev. 7, "Thimble Assembly - Flux Bottom Mounted Instrumentation."

42. ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, 1995 Edition with 1996 Addenda.

43. TMM-015, Rev. 34, "Inservice Repair and Replacement."

APPENDIX A
ILLUSTRATIONS

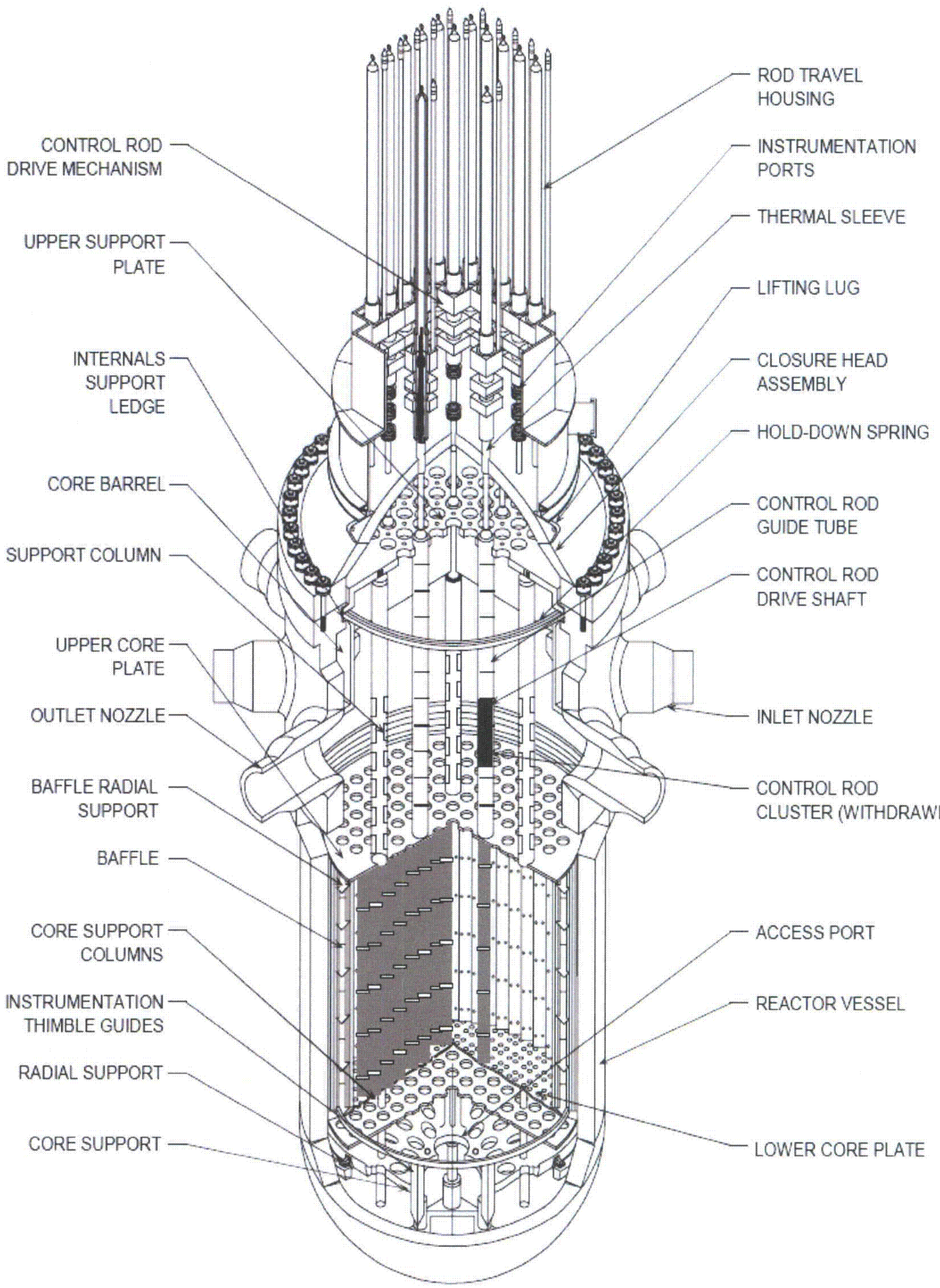

Figure A-1 Illustration of a Typical Westinghouse Internals

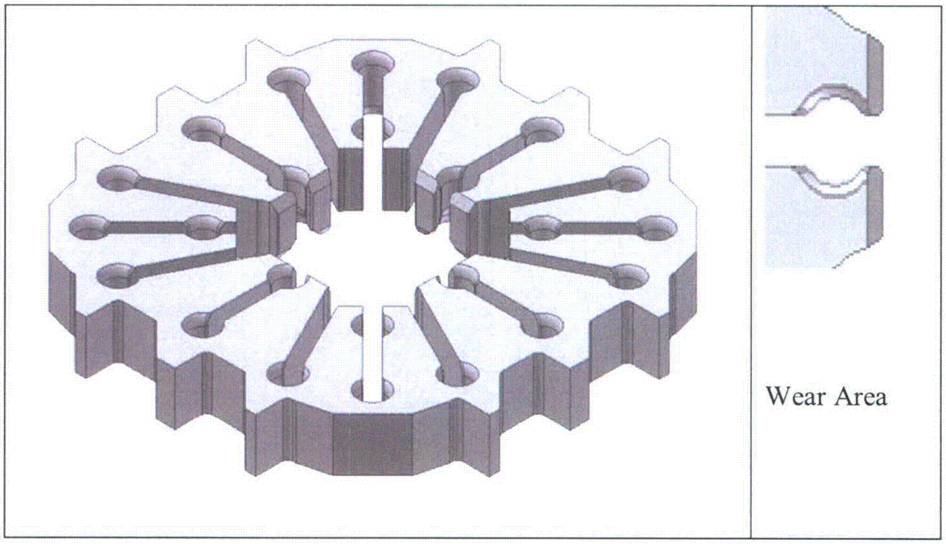

Figure A-2 Typical Westinghouse Control Rod Guide Card

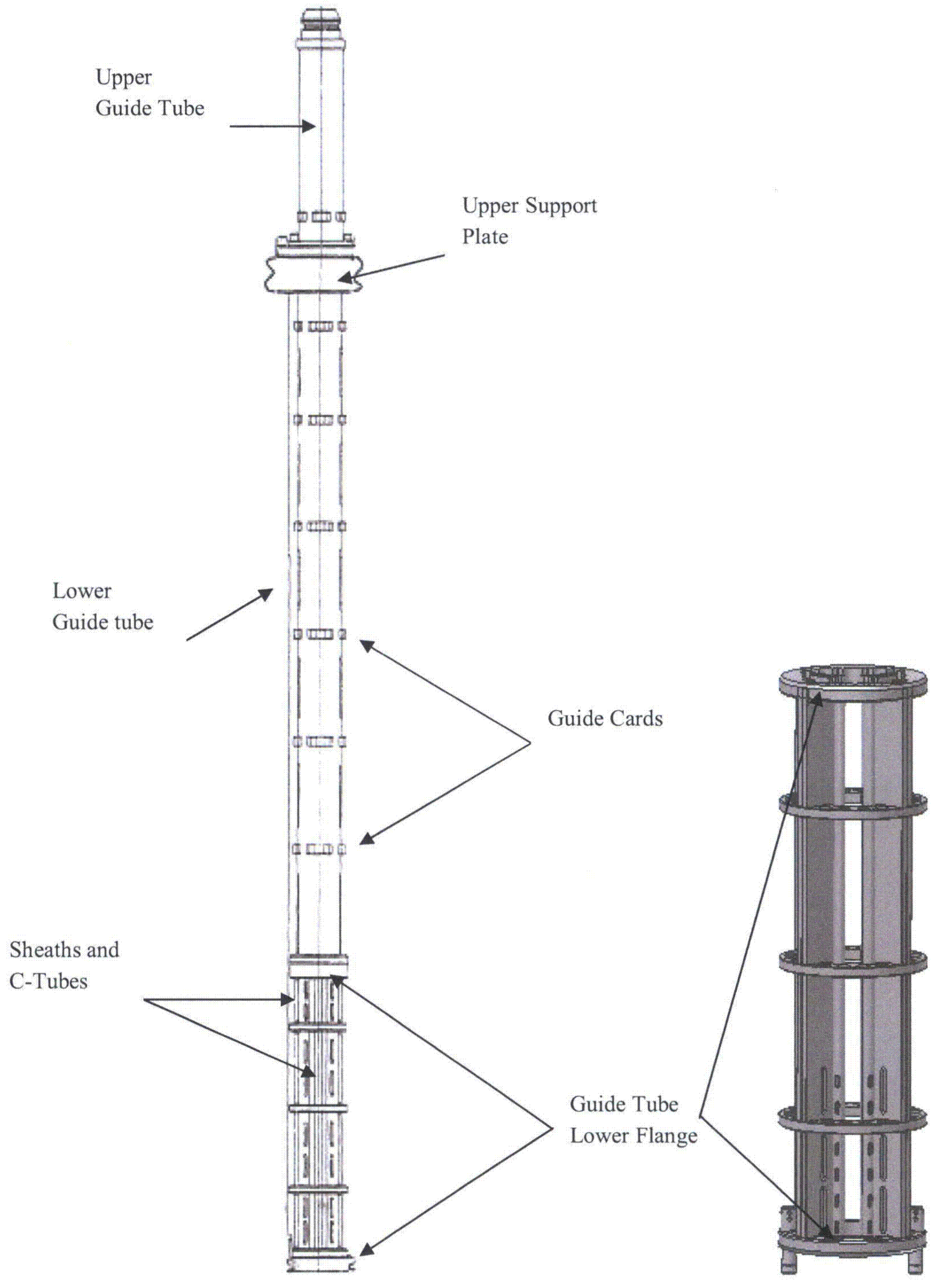

Figure A-3 Lower Section of Control Rod Guide Tube Assembly

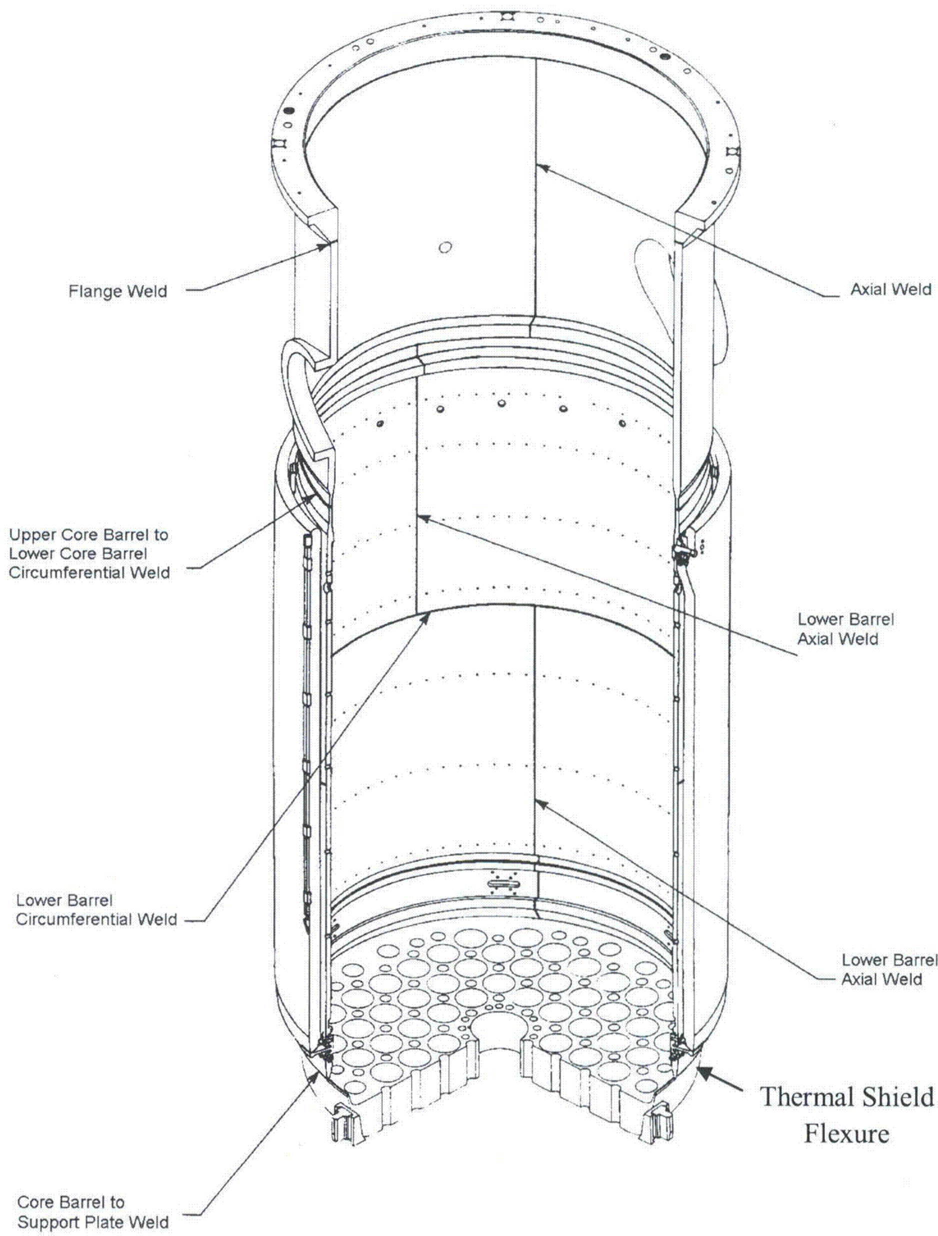

Figure A-4 Major Core Barrel Welds

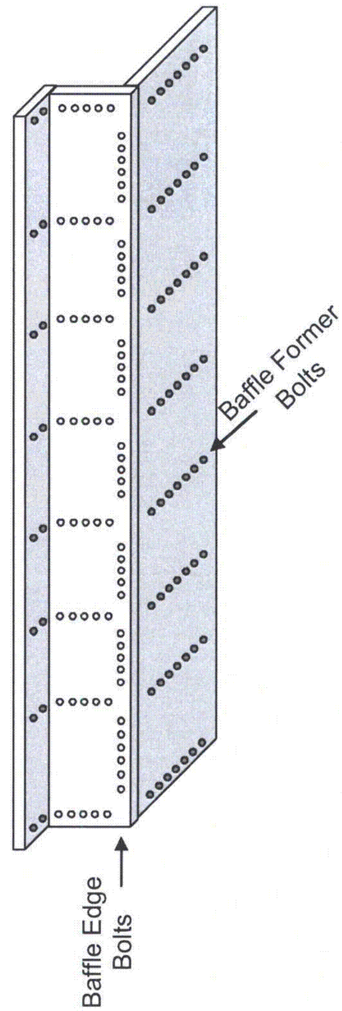

Figure A-5 Bolting Systems used in Westinghouse Core Baffles

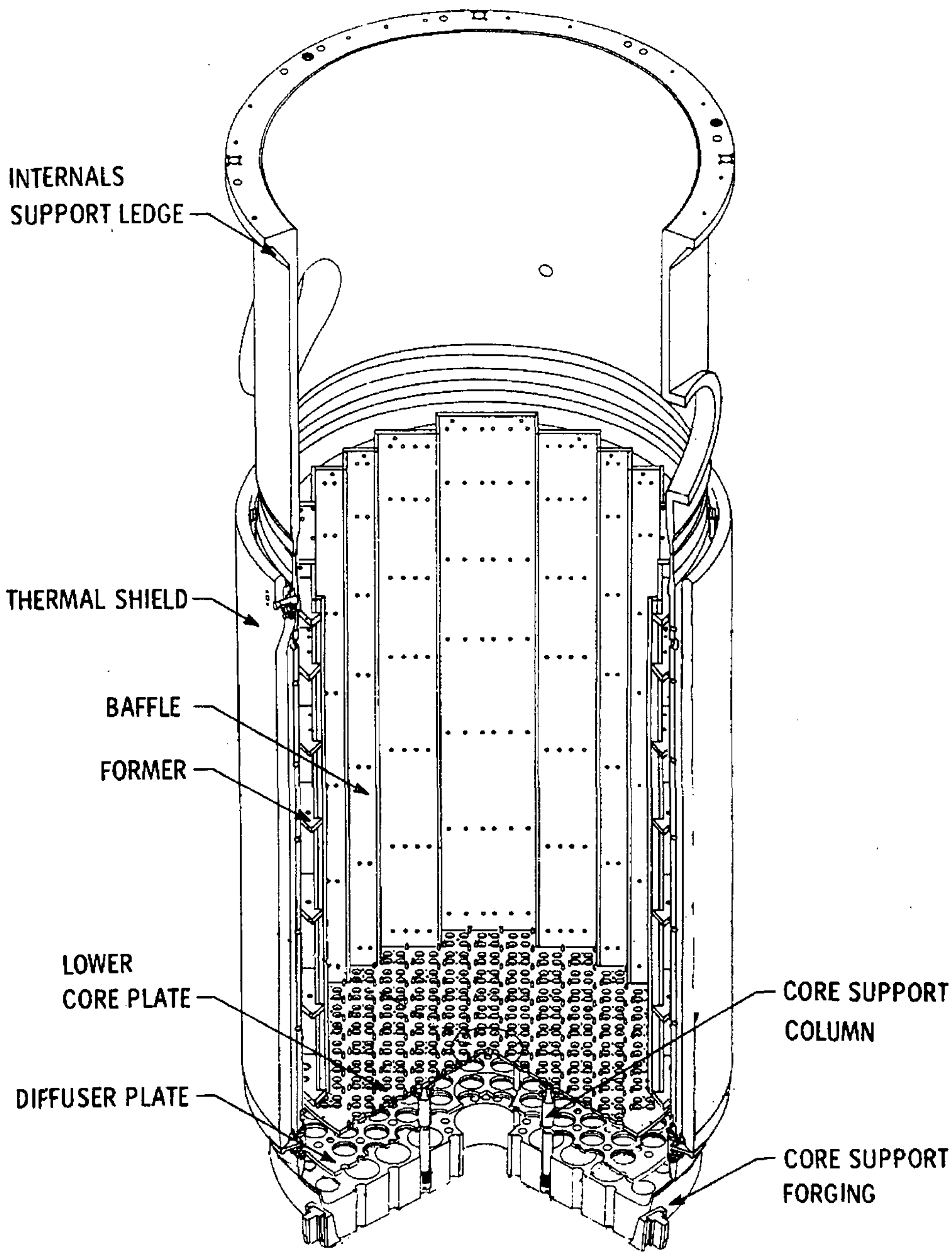

Figure A-6 Core Baffle/Barrel Structure

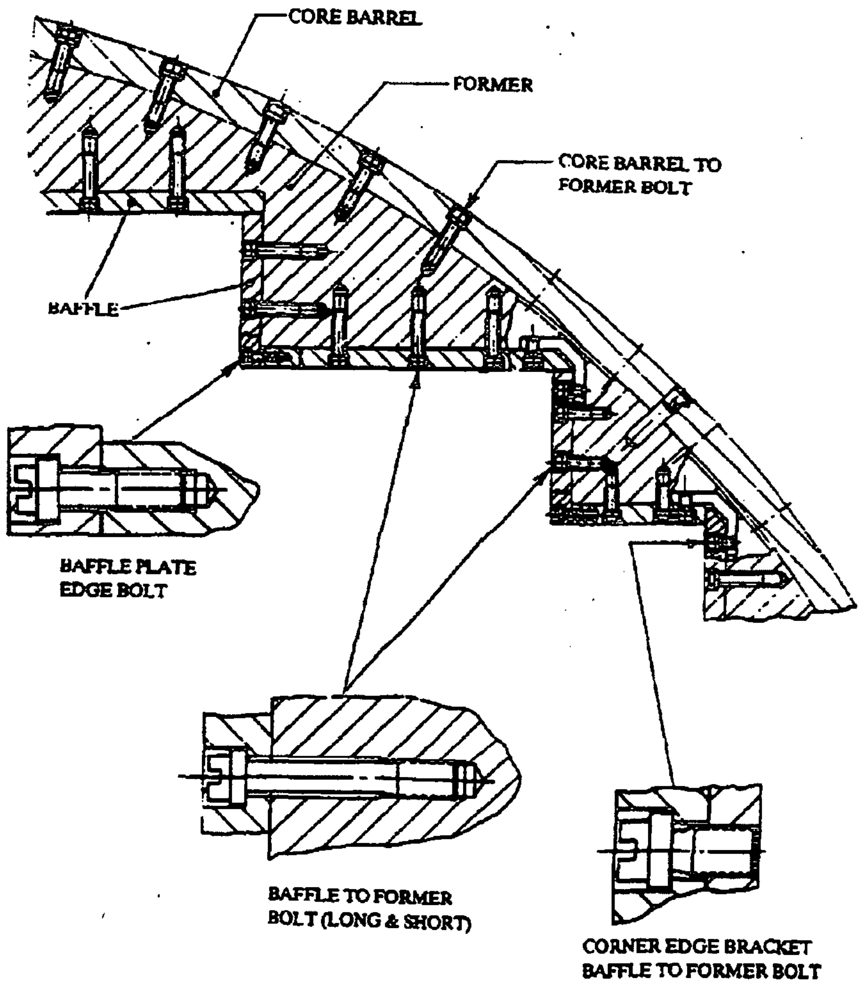

Figure A-7 Bolting in a Typical Westinghouse Baffle-Former Structure

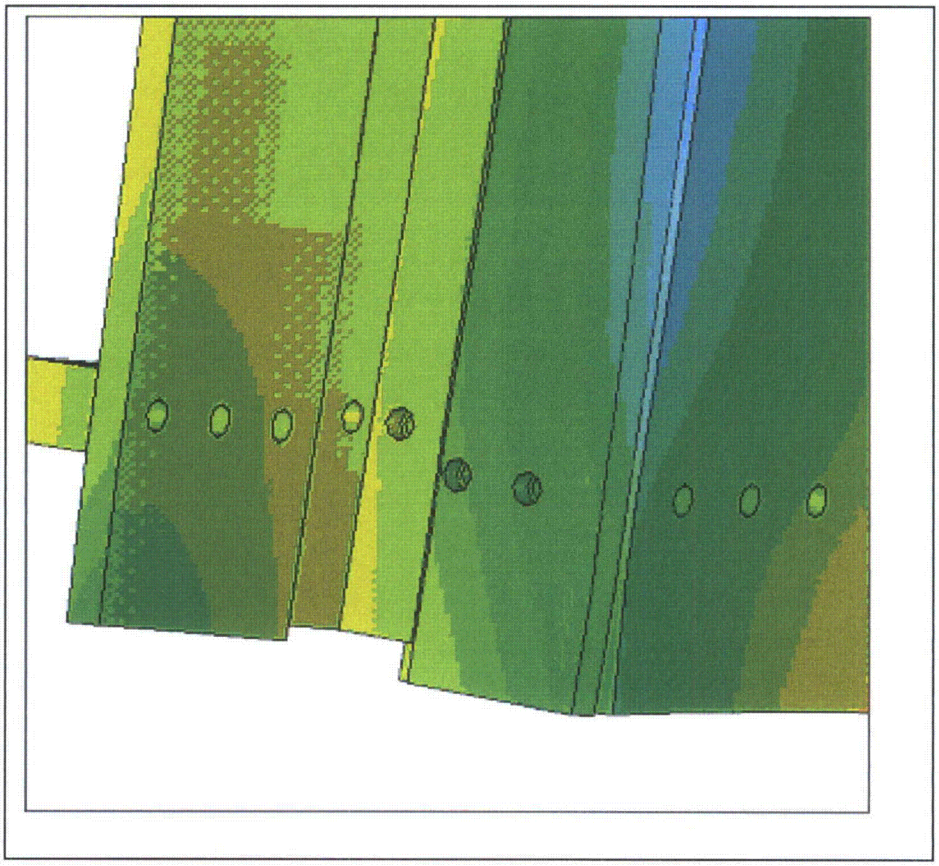

Figure A-8 Vertical Displacement between the Baffle Plates and Bracket at the Bottom of the Baffle-Former-Barrel Assembly

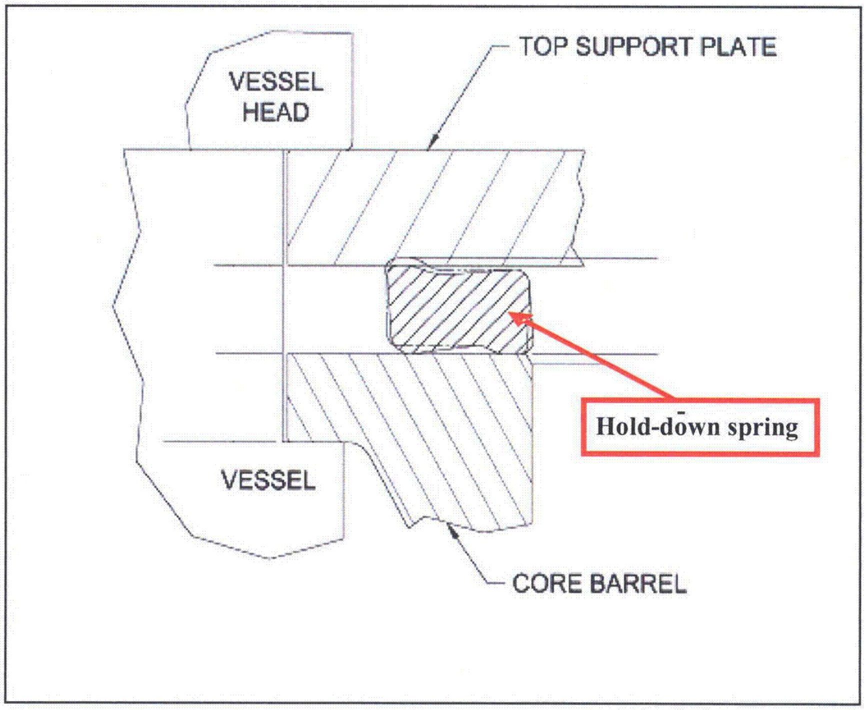

Figure A-9 Schematic Cross-Sections of the Westinghouse Hold-down Springs

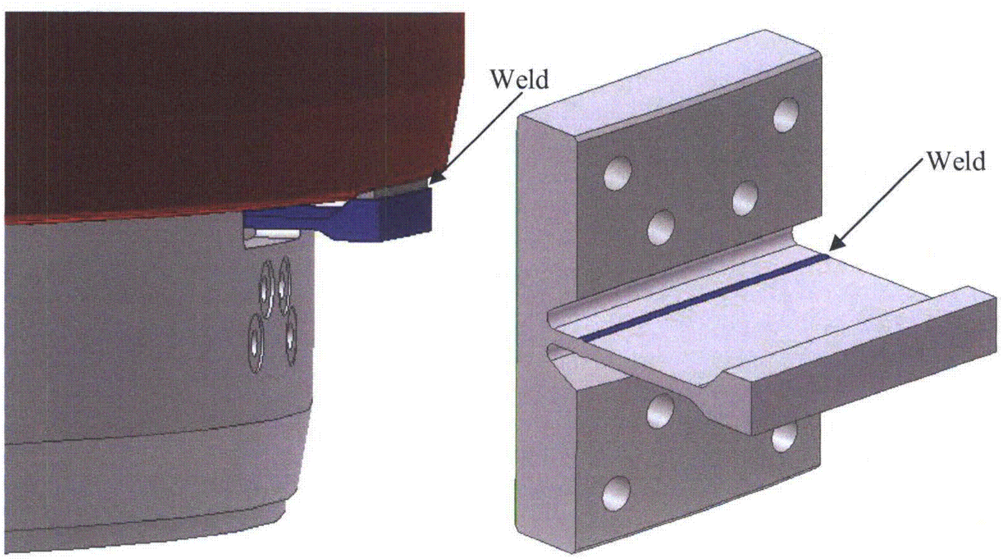

Figure A-10 Typical Thermal Shield Flexure

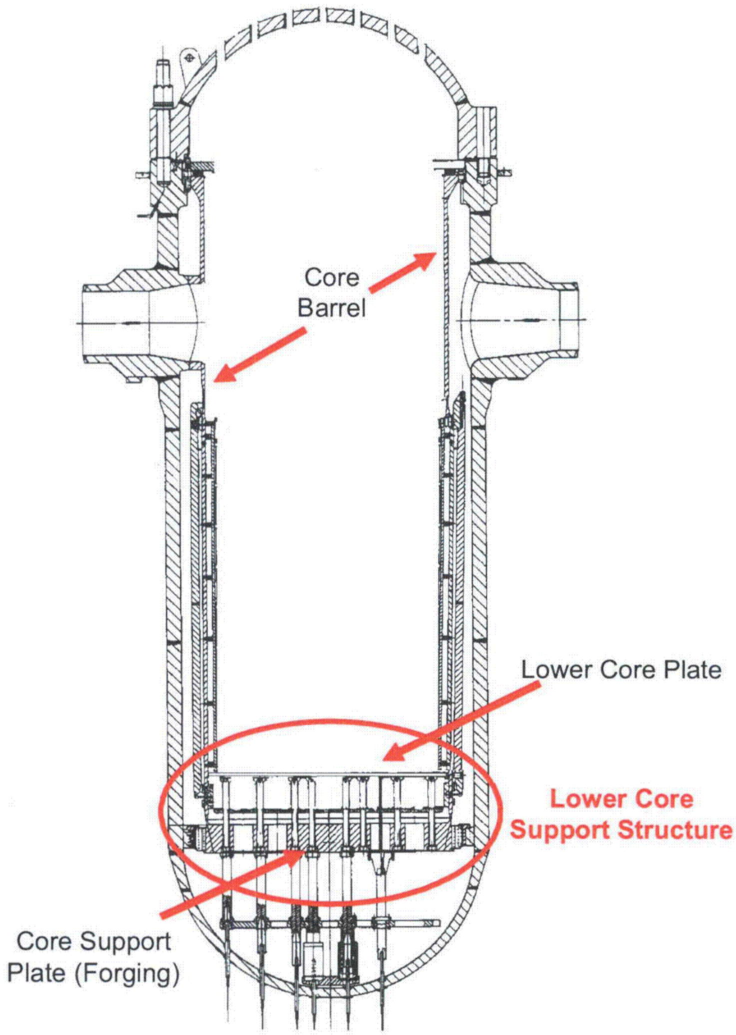

Figure A-11 Lower Core Support Structure

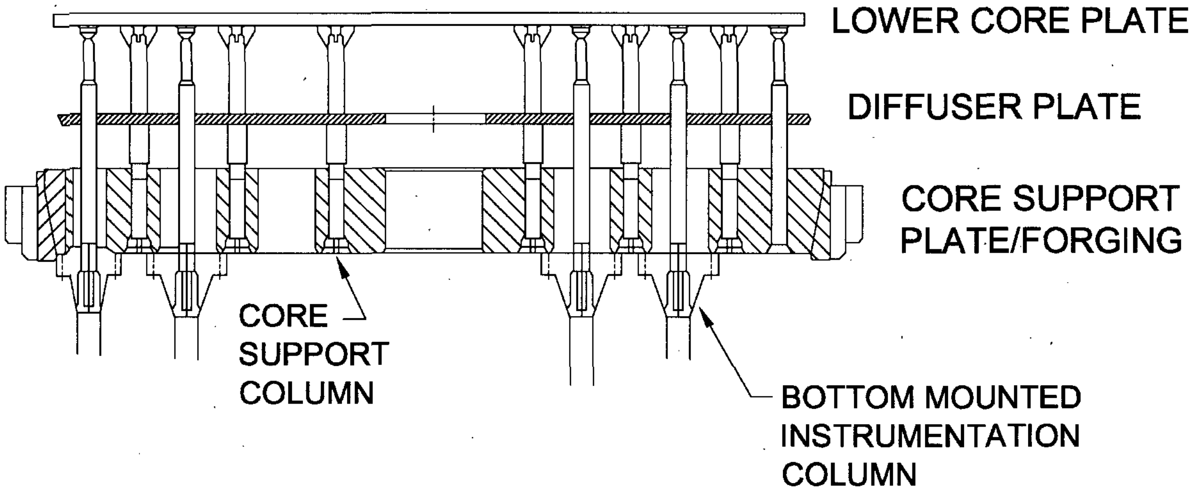

Figure A-12 Lower Core Support Structure – Core Support Plate Cross-Section

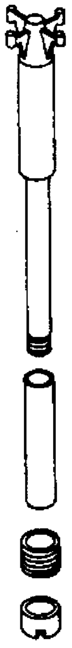

Figure A-13 Typical Core Support Column

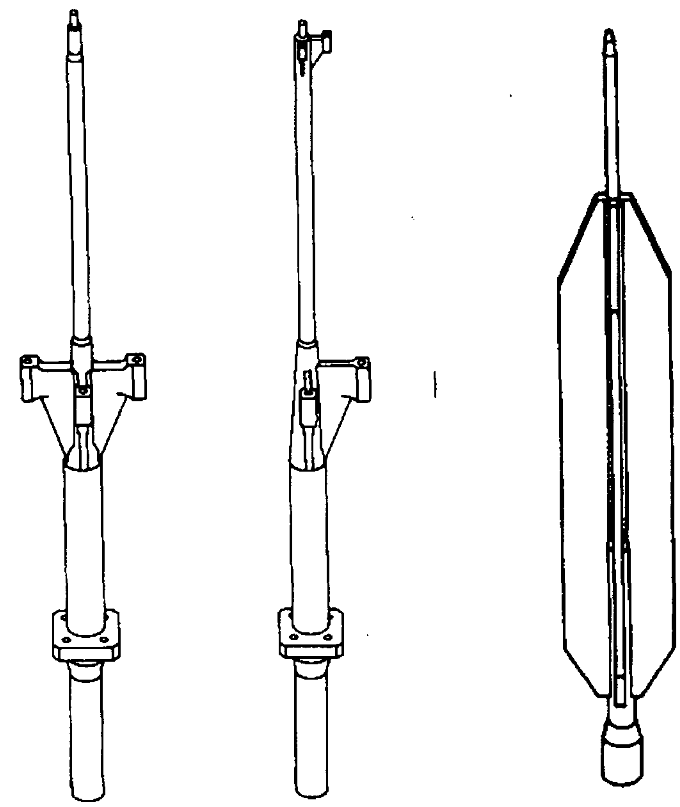

Figure A-14 Examples of BMI Column Designs

APPENDIX B
ROBINSON LICENSE RENEWAL AGING MANAGEMENT REVIEW SUMMARY TABLES

The content and numerical identifiers in Tables B-1 and B-2 of Appendix B are extracted from Table 3.1-1 (AMPs evaluated in the GALL report that are relied on for license renewal) and Table 3.1-2 (evaluations that are different from or not addressed in the GALL report) of the license renewal application approved by the NRC.

Table B-1 LRA Aging Management Review Summary Table 3.1-1 Robinson LRA

Component/Commodity Group[1]	Aging Effect/Mechanism	Aging Management Program	Comments
5. Baffle-Former Assembly Baffle-Former Bolts	Loss of fracture toughness due to neutron irradiation embrittlement and void swelling	PWR Vessel Internals Program	
8. Reactor Internals	Changes in dimension due to void swelling	PWR Vessel Internals Program	
Baffle-Former Assembly 12. Baffle-Former Bolts	Crack initiation and growth due to SCC and IASCC	PWR Vessel Internals Program and Water Chemistry Program	
Baffle-Former Assembly 13. Baffle-Former Bolts	Loss of preload due to stress relaxation	ASME Section XI, Subsection IWB, IWC, and IWD Program and PWR Vessel Internals Program	
25. Reactor Vessel Internals CASS components	Loss of fracture toughness due to thermal aging, neutron irradiation embrittlement, and void swelling	PWR Vessel Internals Program	See also Table B-2, Item 14
28. Reactor Internals, Reactor Vessel Closure Studs, and Core Support Pads	Loss of material due to wear	ASME Section XI, Subsection IWB, IWC, and IWD Program and Flux Thimble Eddy Current Inspection Program	NRC Bulletin 88-09 (Eddy current)
30. Upper and Lower Internals Assembly (upper internals hold-down spring and lower internals assembly clevis insert bolts)	Loss of preload due to stress relaxation	ASME Section XI, Subsection IWB, IWC, and IWD Program and PWR Vessel Internals Program	See also Table B-2, Item 15
31. Reactor Vessel Internals in Fuel Zone Region (except baffle bolts)	Loss of fracture toughness due to neutron irradiation embrittlement and void swelling	PWR Vessel Internals Program and Primary Water Chemistry	
33. Reactor Vessel Internals (except baffle former bolts)	Crack initiation and growth due to SCC and IASCC	PWR Vessel Internals Program and Primary Water Chemistry	

Table B-1 LRA Aging Management Review Summary Table 3.1-1 Robinson LRA

Component/Commodity Group[1]	Aging Effect/Mechanism	Aging Management Program	Comments
35. Reactor Internals (upper and lower internal assemblies)	Loss of preload due to stress relaxation	ASME Section XI, Subsection IWB, IWC, and IWD Program and PWR Vessel Internals Program	See Table B-2, Item 15

Note:

1. The numbers contained in this column reflect the identical numbers in the RNP LRA table referenced.

Table B-2 LRA Aging Management Review Summary Table 3.1-2 Robinson LRA

Component/Commodity Group(1)	Aging Effect/Mechanism	Aging Management Program	Comments
14. Reactor Vessel Internals CASS Components	Reduction of fracture toughness from thermal embrittlement and neutron irradiation embrittlement	PWR Vessel Internals Program	
15. Reactor Internals: Upper Support Column Bolts, Hold-down Spring, Lower Support Plate Column Bolts, and Clevis Insert Bolts	Loss of preload due to stress relaxation	ASME Section XI, Subsection IWB, IWC, and IWD Program and PWR Vessel Internals Program	
16. Flux Thimbles	Cracking from SCC	Primary Water Chemistry	

Note:

1. The numbers contained in this column reflect the identical numbers in the RNP LRA table referenced.

APPENDIX C
MRP-227 AUGMENTED INSPECTIONS

Table C-1 MRP-227 Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Control Rod Guide Tube Assembly Guide plates (cards)	All plants	Loss of material (wear)	None	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period, and no earlier than two refueling outages prior to the start of the license renewal period. Subsequent examinations are required on a ten-year interval.	20% examination of the number of CRGT assemblies, with all guide cards within each selected CRGT assembly examined(1) See Figure A-2.
Control Rod Guide Tube Assembly Lower flange welds	All plants	Cracking (SCC, fatigue)	Bottom-mounted instrumentation (BMI) column bodies, lower support column bodies (cast)	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval	100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal See Figure A-3.
Core Barrel Assembly Upper core barrel flange weld	All plants	Cracking (SCC)	Remaining core barrel welds, Lower support column bodies (non-cast)	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval	100% of one side of the accessible surfaces of the selected weld and adjacent base metal See Figure A-4.

Table C-1 MRP-227 Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Baffle-Former Assembly Baffle-edge bolts	All plants with baffle-edge bolts	Cracking (IASCC, fatigue) that results in: Lost or broken locking devices Failed or missing bolts Protrusion of bolt heads	None	Visual (VT-3) examination, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval	Bolts and locking devices on high-fluence seams. 100% of components accessible from core side See Figures A-5, A-6, and A-7
Baffle-Former Assembly Baffle-former bolts	All plants	Cracking (IASCC, Fatigue)	Lower support column bolts, barrel-former bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination after 10 to 15 additional EFPY to confirm stability of bolting pattern. Re-examination for high-leakage core designs requires continuing examinations on a ten-year interval.	100% of accessible bolts or as supported by plant-specific justification. Heads accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs. See Figures A-5 and A-6

Table C-1 MRP-227 Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Baffle-Former Assembly Assembly	All plants	Distortion (void swelling), or cracking (IASCC) that results in: Abnormal interaction with fuel assemblies Gaps along high-fluence baffle joint Vertical displacement of baffle plates near high-fluence joint Broken or damaged edge bolt locking systems along high-fluence baffle joints	None	Visual (VT-3) examination to check for evidence of distortion, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval	Core side surface, as indicated See Figure A-8

Table C-1 MRP-227 Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Alignment and Interfacing Components Internals hold-down spring	All plants with 304 stainless steel hold-down springs	Distortion (loss of load) Note: This mechanism was not strictly identified in the original list of age-related degradation mechanisms (ARDM).	None	Direct measurement of spring height within three cycles of the beginning of the license renewal period. If the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages, in order to extrapolate the expected spring height to 60 years	Measurements should be taken at several points around the circumference of the spring, with a statistically adequate number of measurements at each point to minimize uncertainty. Replacement of 304 springs by 403 springs is required when the spring stiffness is determined to relax beyond design tolerance. See Figure A-9
Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields	Cracking (fatigue) or loss of material (wear) that results in thermal shield flexures excessive wear, fracture, or complete separation	None	Visual (VT-3) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval	100% of thermal shield flexures See Figures A-6 and A-10

Note:
1. The 20% figure was determined by majority consensus of the MRP RI-FG members. An assessment will be performed upon receipt of the recommendations of the PWROG study when available.

Table C-2 MRP-227 Expansion Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Core Barrel Assembly Barrel-former bolts	All plants	Cracking (IASCC, fatigue)	Baffle-former bolts	Volumetric (UT) examination, with initial and subsequent examinations dependent upon results of baffle-former bolt examinations	100% of accessible bolts. Accessibility may be limited by presence of thermal shields or neutron pads. See Figure A-7
Lower Support Assembly Lower support column bolts	All plants	Cracking (IASCC, fatigue)	Baffle-former bolts	Volumetric (UT) examination, with initial and subsequent examinations dependent upon results of baffle-former bolt examinations	100% of accessible bolts or as supported by plant-specific justification See Figures A-11, A-12 and A-13
Core Barrel Assembly Core barrel flange, core barrel outlet nozzles, lower core barrel flange weld	All plants	Cracking (SCC, fatigue)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination, with initial examination and re-examination frequency dependent upon the examination results for upper core barrel flange	100% of one side of the accessible surfaces of the selected weld and adjacent base metal See Figure A-6
Lower Support Assembly Lower support column bodies (non-cast)	All plants	Cracking (IASCC)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination, with initial examination and re-examination frequency dependent upon the examination results for upper core barrel flange weld	100% of accessible surfaces See Figures A-11, A-12, and A-13
Lower Support Assembly Lower support column bodies (cast)	All plants	Cracking (IASCC) including the detection of fractured support columns	Control rod guide tube (CRGT) lower flanges	Visual (EVT-1) examination	100% of accessible support columns See Figures A-11, A-12, and A-13

Table C-2 MRP-227 Expansion Inspection and Monitoring Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Bottom-Mounted Instrumentation System Bottom-Mounted Instrumentation (BMI) column bodies	All plants	Cracking (fatigue) including the detection of completely fractured column bodies	Control rod guide tube (CRGT) lower flanges	Visual (VT-3) examination of BMI column bodies as indicated by difficulty of insertion/withdrawal of flux thimbles. Flux thimble insertion/withdrawal to be monitored at each inspection interval	100% of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal See Figures A-12 and A-14

Table C-3 MRP-227 Existing Inspection and Aging Management Programs Credited in Recommendations for Westinghouse-Designed Internals

Item	Applicability	Effect (Mechanism)	Expansion Link[1]	Examination Method/Frequency[1]	Examination Coverage
Core Barrel Assembly Core barrel flange	All plants	Loss of material (wear)	ASME Code Section XI	Visual (VT-3) examination to determine general condition for excessive wear	All accessible surfaces at specified frequency
Upper Internals Assembly Upper support ring or skirt	All plants	Cracking (SCC, fatigue)	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces at specified frequency
Lower Internals Assembly Lower core plate XL[1] lower core plate	All plants	Cracking (IASCC, fatigue)	ASME Code Section XI	Visual (VT-3) examination of the lower core plates to detect evidence of distortion and/or loss of bolt integrity	All accessible surfaces at specified frequency
Lower Internals Assembly Lower core plate XL[1] lower core plate	All plants	Loss of material (wear)	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces at specified frequency
Bottom-Mounted Instrumentation System Flux thimble tubes	All plants	Loss of material (wear)	NUREG-1801, Rev. 1	Surface (ET) examination	Eddy current surface examination, as defined in plant response to IEB 88-09
Alignment and Interfacing Components Clevis insert bolts	All plants	Loss of material (wear)[2]	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces at specified frequency
Alignment and Interfacing Components Upper core plate alignment pins	All plants	Loss of material (wear)	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces at specified frequency

Notes:
1. XL = "Extra Long," referring to Westinghouse plants with 14-foot cores.
2. Bolt was screened-in because of stress relaxation and associated cracking; however, wear of the clevis/insert is the issue.

Table C-4 MRP-227 Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link[(1)]	Expansion Criteria	Additional Examination Acceptance Criteria
Control Rod Guide Tube Assembly Guide plates (cards)	All plants Figure A-2	Visual (VT-3) Examination The specific relevant condition is wear that could lead to loss of control rod alignment and impede control assembly insertion.	None	Not applicable	Not applicable
Control Rod Guide Tube Assembly Lower flange welds	All plants Figure A-3	Enhanced visual (EVT-1) examination The specific relevant condition is a detectable crack-like surface indication.	a. Bottom-mounted instrumentation (BMI) column bodies b. Lower support column bodies (cast)	a. Confirmation of surface breaking indications in two or more CRGT lower flange welds, combined with flux thimble insertion/withdrawal difficulty, shall require visual (VT-3) examination of BMI column bodies by the completion of the next refueling outage. b. Confirmation of surface breaking indications in two or more CRGT lower flange welds shall require EVT-1 examination of cast lower support column bodies within three fuel cycles following the initial observation.	a. For BMI column bodies, the specific relevant condition for the VT-3 examination is completely fractured column bodies. b. For cast lower support column bodies, the specific relevant condition is a detectable crack-like surface indication.

Table C-4 MRP-227 Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link[1]	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Upper core barrel flange weld	All plants Figure A-4	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Remaining core barrel welds Lower support column bodies (non cast)	a. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel flange weld shall require that the EVT-1 examination, and any supplementary UT examination, be expanded to include the core barrel-to-support plate weld by the completion of the next refueling outage. If extensive confirmed indications in the core barrel-to-support plate weld are detected, further expansion of the EVT-1 examination shall include the remaining core barrel assembly welds. b. If extensive cracking in the remaining core barrel welds is detected, EVT-1 examination shall be expanded to include the upper six inches of the accessible surfaces of the non-cast lower support column bodies within three fuel cycles follow the initial observation.	a and b. The specific relevant condition is a detectable crack-like surface indication.

Table C-4 MRP-227 Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(1)	Expansion Criteria	Additional Examination Acceptance Criteria
Baffle-Former Assembly Baffle-edge bolt	All plants with baffle-edge bolts Figures A-5, A-6, and A-7	Visual (VT-3) examination. The specific relevant conditions are missing or broken locking devices, failed or missing bolts, and protrusion of bolt heads.	None	Not applicable	Not applicable
Baffle-Former Assembly Baffle-former bolts	All plants Figures A-5 and A-6	Volumetric (UT) examination. The examination acceptance criteria for the UT of the baffle-former bolts shall be established as part of the examination technical justification.	a. Lower support column bolts b. Barrel-former bolts	a. Confirmation that more than 5% of the baffle-former bolts actually examined on the four baffle plates at the largest distance from the core (presumed to be the lowest dose locations) contain unacceptable indications shall require UT examination of the lower support column bolts within the next three fuel cycles. b. Confirmation that more than 5% of the lower support column bolts actually examined contain unacceptable indications shall require UT examination of the barrel-former bolts.	a and b. The examination acceptance criteria for the UT of the lower support column bolts and the barrel-former bolts shall be established as part of the examination technical justification.

Table C-4 MRP-227 Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals					
Item	**Applicability**	**Examination Acceptance Criteria (Note 1)**	**Expansion Link(1)**	**Expansion Criteria**	**Additional Examination Acceptance Criteria**
Baffle-Former Assembly Assembly	All plants Figure A-8	Visual (VT-3) examination. The specific relevant conditions are evidence of abnormal interaction with fuel assemblies, gaps along high-fluence shroud plate joints, vertical displacement of shroud plates near high-fluence joints, and broken or damaged edge bolt locking systems along high fluence baffle plate joints.	None	Not applicable	Not applicable

Table C-4 MRP-227 Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link[1]	Expansion Criteria	Additional Examination Acceptance Criteria
Alignment and Interfacing Components Internals hold-down spring	All plants with 304 stainless steel hold-down springs NOTE: RNP hold-down spring is 304SS Figure A-9	Direct physical measurement or spring height. The examination acceptance criterion for this measurement is that the remaining compressible height of the spring shall provide hold-down forces within the plant-specific design tolerance.	None	Not applicable	Not applicable
Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields Figures A-6 and A-10	Visual (VT-3) examination. The specific relevant conditions for thermal shield flexures are excessive wear, fracture, or complete separation.	None	Not applicable	Not applicable

Note:
1. The examination acceptance criterion for visual examination is the absence of the specified relevance condition(s).

The previous RNP-RA/09-0081 letter, dated September 24, 2009, did not have the enclosure included. The attached document includes the letter and enclosure.